AUDIT SUMMARY FOR THE REGULATORY AUDIT OF THE NUSCALE POWER, LLC, DESIGN CERTIFICATION APPLICATION, EMERGENCY CORE COOLING SYSTEM BORON REDISTRIBUTION ISSUE AND APPLICABLE DESIGN CHANGES

1.0 BACKGROUND

By letter dated December 31, 2016, NuScale Power, LLC (NuScale) submitted to the U.S. Nuclear Regulatory Commission (NRC), a Final Safety Analysis Report (FSAR) for its Design Certification (DC) application of the NuScale small modular reactor (SMR) design (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A229). In early 2020, NuScale informed the NRC staff that, in certain loss-of-coolant accident (LOCA) scenarios, the emergency core cooling system (ECCS) may actuate later than expected. If the decay heat removal system (DHRS) operates as designed, cooling and eventually condensation would occur in the steam generator (SG) region, and dilute condensate could mix with borated water in the downcomer. Meanwhile, the containment vessel (CNV) would fill with condensate. When the ECCS valves open, the liquid in the CNV could enter through the reactor recirculation valves (RRVs) and push cold, deborated water into the core region as ECCS flow is established. Later, NuScale informed the NRC staff that downcomer deboration can similarly result from extended DHRS cooling following a non-LOCA event with riser uncovery. NuScale introduced a series of design changes to address these issues related to boron redistribution and the associated return to power analyses, which affected multiple chapters of the Final Safety Analysis Report (FSAR).

The purpose of this regulatory audit was to: (1) better understand the impact of three submitted design changes on information related to Chapter 3, “Design of Structures, Components, Equipment, and Systems”; Chapter 4, “Reactor”; Chapter 5, “Reactor Coolant System and Connecting Systems”; Chapter 6, “Engineered Safety Features”; Chapter 7, “Instrumentation and Controls”; Chapter 13, “Conduct of Operations”; Chapter 15, “Transient and Accident Analyses”; Chapter 16, “Technical Specifications”; and Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation,” as well as associated technical and topical reports; (2) confirm information through examination of calculations and other information; (3) identify information that would require docketing to support a regulatory finding; and (4) ensure conformance with applicable NRC regulations. The audit entrance meeting was held on March 4, 2020, at the NRC Headquarters located in Rockville, Maryland.

The audit included the NRC staff’s review of docketed and non-docketed information via the NuScale electronic reading room (eRR) and daily workweek audit teleconferences.

During this audit, NuScale provided, in the eRR, FSAR chapter markups and applicable information for three design changes: (1) high containment level ECCS actuation setpoint change; (2) new ECCS actuation signal on low reactor coolant system (RCS) pressure; and (3) addition of riser holes. The NRC staff examined documents and analyses that support the design changes. All of the NRC staff’s questions and requests for clarification were resolved during the audit.

2.0 REGULATORY AUDIT BASIS

- Title 10, “Energy,” of the *Code of Federal Regulations*, Part 50, “Domestic Licensing of Production and Utilization Facilities,” (10 CFR Part 50), Appendix A, “General Design Criteria for Nuclear Power Plants,” and Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

- 10 CFR 50.43, “Additional standards and provisions affecting class 103 licenses and certifications for commercial power,” paragraph (e).

- 10 CFR 50.55a, “Codes and standards.”

Relevant regulatory guidance includes:

- NRC Regulatory Guide (RG) 1.100, Revision 3, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants.”

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition”

3.0 AUDIT LOCATION AND DATES

The audit was conducted from the NRC headquarters, via NuScale’s eRR, and via a teleconference bridge line.

Dates: March 4, 2020, through June 26, 2020

Locations: NRC Headquarters
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NuScale eRR

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Snuggerud, Ross - Chief of Engineering
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6.0 AUDIT DOCUMENTS

The NRC staff audited the following documents provided by NuScale:

Boron Redistribution Issue eRR Document List	
Document Number	**Document Title**
ER-0000-2486	Safety Analysis Limits Report, Revision 8
EE-0000-7939	Small Break LOCA Boron Dilution Analysis, Revision 0
ECN-0000-8182	Leakage Boron Dilution Analysis, Revision 0
N/A	RVV and RRV Passive Opening Time Differences due to Hydrostatic Head
N/A	Additional Boron Transport Results for Leakage Cases
ECN-A023-8056	Riser flow holes update, Revision 0
ECN-A010-8071	Design Change Impact Analysis, Revision 0
ECN-0000-7963	Design Change Impact Analysis, Revision 0
ECN-A010-8054	FIV Screening of Riser Flow Holes, Revision 0
ECN-E011-7937	Modify MPS Single Failure Analysis to address boron dilution issue, Revision 0
ECN-E000-7934	D3 Assessment Update for New ECCS Logic, Revision 0
ECN-A013-7888	Additions to Evaluate Impact of Change in ECCS Actuation Signals and Inclusion of Riser Holes, Revision 0

ECN-A013-7888	Additional SLB and FWLB Plots
ECN-A030-8055	Add Loss for Upper Riser Holes, Revision 0
ECN-A023-8125	Assessment of Acoustic Noise due to Riser Flow Holes, Revision 0
ECN-A023-8122	Riser Holes Turbulent Buffeting Assessment, Revision 0
ECN-0000-8080	Assessment of Riser Holes Impact on Non-LOCA Events, Revision 0
ECN-0000-8053	Digital-Based Common Cause Failure of Wide-Range RCS Pressure Signal Coping Analysis, Revision 0
EC-A030-2359	RCS Flow Form Loss Calculation, Revision 5
EC-A023-3535	RVI Turbulent Buffeting Degradation Evaluation, Revision 1
EC-A010-1782	Reactor Module NRELAP5 Model, Revision 3
EC-0000-2749	Loss of Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks, Revision 3
EC-0000-7929	Evaluation of Boron Transport during Extended DHRS Operation
ER-0000-6621	Boron Transport and Distribution Methodology, Revision 1
EC-0000-7110	Extended ECCS Transient for Boron Dilution Analysis, Revision 0
EC-0000-7127	Boron Dilution Analysis for IORV (RRV) Followed by ECCS Actuation, Revision 0
EC-0000-7144	Boron Dilution Analysis for IORV (RVV) Followed by ECCS Actuation, Revision 0
ER-P000-8072	PRA Impact Analysis of Boron Redistribution Design Changes
ER-P000-8076	PRA Thermal-Hydraulic Impact Analysis of Boron Redistribution Design Changes
ER-B020-3817	Proof of Concept Test Report for Emergency Core Cooling (ECC) System Valves, Revision B
EQ-B020-2140	ASME Design Specification for Emergency Core Cooling System Valves, Revision 3
CP-2066	Approved LTC TR Change Package for Steam Space Break Disposition
CP-2075	Final FSAR, Chapter 3 Markups
CP-2076	Final FSAR, Chapter 6
CP-2055	Final FSAR, Chapter 7
CP-2047	Final FSAR, Chapter 16
CP-2077	Final FSAR, Chapter 17
CP-2082; CP-2091	Final FSAR, Chapter 19
CP-2074	Final Tier 1, Chapter 2 Markups
CP-2056	Final TR-0316-22048 Revision 3 Markups
CP-2068	Final TR-0516-49084 Revision 3 Markups
CP-2057	Final TR-0616-49121 Revision 3 Markups
CP-2047	Final TR-1116-52011 Revision 4 Redline
CP-2092	Final FSAR, Chapter 5
CP-2050	Final FSAR, Chapter 15
CP-2071	Final FSAR, Chapter 4 and 15
CP-2083	LOCA topical report changed pages
CP-2107	Draft Change Package – LOCA Topical Report
CP-2084	Non-LOCA topical report changed pages
	Generic Technical Specifications Volume 1: Specifications
	Generic Technical Specifications Volume 2: Bases

CR-0220-69077	NPM-160 steam space LOCA progression and ECCS actuation
CR-0120-68681	Disposition downcomer boron concentration from extended DHRS cooling

7.0 DESCRIPTION OF AUDIT ACTIVITIES AND SUMMARY OF OBSERVATIONS

The NRC staff audited information related to boron redistribution and the associated design changes and return to power analyses that is associated with multiple chapters of the FSAR. Specific observations for each chapter and topical report are documented below.

Chapter 3 - Design of Structures, Components, Equipment, and Systems **(NRR/DEX/EMIB)**

With respect to mechanical aspects, the resolution of the boron redistribution issue for the NuScale SMR involved: (1) the incorporation of holes through the upper riser of the NuScale Nuclear Power Module (NPM), and (2) the assumption of the self-opening of the ECCS main valves by their internal springs at very low differential pressure conditions. These two mechanical aspects are addressed below:

Riser Holes

NuScale added four small through holes to the NPM upper riser as part of its design changes to address the boron redistribution issue. The holes permit flow between the upper riser and SG fluid regions during certain DHRS conditions where the height of the fluid in the upper riser can be below that of the riser itself. NuScale evaluated several flow-induced vibration (FIV) mechanisms caused by the presence of the holes at normal operating conditions.

The NRC staff evaluated the following documents for this audit:

- Engineering Change Notice ECN-A023-8125, “Assessment of Acoustic Noise due to Riser Flow Holes,” Revision 2, May 15, 2020.
- ECN-A023-8122, “Riser Holes Turbulent Buffeting Assessment,” Revision 1, May 14, 2020.
- ECN-A010-8054, “FIV Screening of Riser Flow Holes,” Revision 1, April 30, 2020.
- ECN-A023-8056, “Riser flow holes update,” Revision 2, April 29, 2020.
- Proposed changes to FSAR Section 3.9.5 on page 3.9-48.

Also, a teleconference was held between NuScale and the NRC staff on May 6, 2020, to discuss the incorporation of the riser holes in the NuScale design, and the quality assurance (QA) process in evaluating the riser holes and their potential impact.

NuScale considered several phenomena with respect to the potential impact of the riser holes:

- flow instabilities due to shear flow passing over the holes (these are the same instabilities that are evaluated for steam lines with side branches),

- whether those flow instabilities, should they occur, might be amplified by acoustic resonances,

- effects of the jets discharging the holes which impinge on neighboring SG tubes, and

- effects of the holes on the structural vibrations and alternating stresses in the upper riser shell induced by turbulent buffeting (TB).

NuScale relied on well-established references to evaluate the relative importance of these phenomena, including Blevins, “Flow Induced Vibration,” 2nd Edition, 1990; and Naudascher and Rockwell, “Flow-Induced Vibrations - an Engineering Guide,” 1994.

<u>Shear Layer Flow Instabilities and Acoustic Resonances</u>

The most significant issue for the evaluation of the riser holes is that of shear layer flow instabilities over open holes. The frequencies of these instabilities may be calculated using

$$f_n = \frac{0.33\left(n - \frac{1}{4}\right)U}{L}$$

where n is the instability harmonic (1, 2, …), U is the flow speed over the hole, and L is the hole diameter. For a hole diameter of ¾ inch, a flow speed of [[ ]] in the upper riser (from previous NuScale calculations), the first two instability frequencies are 10.6 [[ ]]. The instability frequencies due to flow on the SG side, although not evaluated by NuScale, would be similar.

The sound radiated solely by these flow instabilities is unlikely to induce significant structural vibration. However, if these instabilities are amplified by other acoustic resonances, the source strength can increase significantly. The most common acoustic mode to couple with a flow instability is that within the hole cavity itself. Since the cavity depth is small [[( )]] and is open to the fluid on the opposite side, a Helmholtz Resonator exists, where the mass within the hole oscillates at a frequency dependent on the effective stiffness of the fluid volume outside the hole. The Helmholtz Resonator frequency for a hole through a plate is

$$f_r = \frac{c}{2\pi}\sqrt{\frac{A}{V l_e}}, \quad l_e = l + 0.96\sqrt{A}$$

where c is the acoustic speed of sound in the coolant, A is the hole area, V is the volume of fluid backing the hole (in this case, inside the SG cavity or the riser cavity), and le is the effective depth of the hole. For holes backed by large bodies of fluid, the effective depth is longer than just the hole thickness (for the [[ ]] diameter holes, the effective depth is [[ ]]). NuScale did not evaluate the Helmholtz Resonance frequencies, and did not provide estimates of effective backing volumes. However, the NRC staff estimated that the volume required for a Helmholtz resonator frequency to be coincident with the first fluid instability frequency is about 60 cubic feet. The reactor pressure vessel (RPV) total approximate volume (using [[ ]]) is over 4,000 cubic feet; therefore, the potential exists for a Helmholtz resonance to be within the range of the flow instability frequencies.

There are also acoustic modes within the riser fluid volume. The lowest frequency mode is associated with a half acoustic wavelength in the vertical direction, which for a [[ ]] with hot pressurized water is about 25 Hz. While this frequency might coincide with the second fluid instability over the holes, those instabilities will generate acoustic pulsations radial to the riser wall, and cannot couple strongly to an acoustic mode with velocity fluctuations oriented in the vertical direction. NuScale instead evaluated the primary acoustic mode across the diameter of the riser fluid volume using:

$$f_r = \frac{c\lambda_j}{\pi D}$$

Since λ_1 is 1.841 (dimensionless frequency parameter associated with the fundamental diametrical acoustic mode of a cylinder), the fundamental acoustic frequency in the radial direction is [[ ]], which is well above the instability frequency. NuScale did not evaluate the lowest acoustic resonance frequency in the SG annulus, but since that annulus is narrower than the upper riser fluid volume, that frequency will be even higher. Therefore, no significant coupling between flow instabilities and the fundamental acoustic resonances in the NPM is expected.

Through Hole Flow Effects

Although the NRC staff cannot rule out the possibility of a Helmholtz resonance coinciding with a flow instability frequency, NuScale has determined that there will be a positive flow rate through the hole (from the upper riser into the SG annulus) at normal operating conditions. The mean velocity of this through-flow is estimated to be about [[ ]], which is over twice that of the flow upward through the riser and downward through the SG annulus. Therefore, it is nearly impossible for a flow instability to develop over the through holes. The NRC staff finds that it is highly unlikely that a significant flow instability could form and be amplified by a Helmholtz or any other resonator.

The through flow, however, does introduce the possibility of a discharge jet impinging on the SG tubes near the holes. For a [[ ]] flow rate, NuScale estimated both the force induced by the jet, as well as the jet's characteristic frequency of oscillation.

The jet oscillation frequency is given by Naudascher and Rockwell as

$$(n + \varepsilon)\frac{U_{jet}}{L}$$

where n = 1, 2, 3,… and ε depends on flow conditions (varies between 0 and 0.5). A conservative (lower bound) frequency uses $\varepsilon = 0$, which gives a jet oscillation frequency over [[ ]]. This frequency is well above that of the fundamental flexural modes of the SG tubing between tubing supports.

The static force imparted by a jet is:

$$A\rho U_{jet}^2,$$

which NuScale estimated as about [[ ]]. Oscillating forces will be even lower than the static force. NuScale did not apply this force to the SG tubes, but instead

showed that this force is more than order of magnitude lower than other forces already considered in previous tubing forced response calculations. Also, the NRC staff believes that while the jet velocity is twice that of the flow downward through the SG region, the mass flow through the hole is very small, and the jet will, in all likelihood, be diffused into the downward flow through the SG. Since the NRC staff has determined that the jet will likely be diffused, and the jet frequency is well above any SG tubing flexural resonance frequencies, the localized jet forces are not expected to cause tube damage.

Effects of Holes on Turbulent Buffeting Induced Riser Vibration

NuScale did not update its calculations of TB induced riser vibration. However, NuScale did make the following qualitative arguments that the riser holes will not lead to TB-induced damage.

- The holes are much smaller than the riser itself ([[ ]] hole diameter vs. [[ ]]), and therefore do not significantly affect the physical or material properties of the riser shell (which is [[ ]] thick).

The NRC staff finds that the riser holes will introduce stress concentrations (generally assumed to be a factor of 3). However, the TB induced vibration in an NPM is extremely small, and the stress concentrations are highly unlikely to lead to alternating stresses which exceed material fatigue limits.

ECCS Main Valve Self-Opening

In the evaluation of potential reactor events that might impact boron redistribution, NuScale considered the ECCS main valves opening by their small internal spring at very low differential pressure conditions between the RPV and the CNV. On May 27, 2020, the NRC staff conducted an audit discussion with NuScale regarding this assumption. During the discussion, the NRC staff obtained information regarding NuScale's actions to demonstrate the performance of the ECCS main valve spring to support the NuScale DC, and subsequent activities for qualification and testing of the ECCS valve system, including the main valve spring. As part of the NuScale design certification application (DCA) review, the NRC staff has evaluated other aspects of the ECCS valve system, including design demonstration testing of its overall performance to satisfy 10 CFR 50.43(e) for this first-of-a-kind design feature for the NuScale SMR, as documented in an audit report dated December 19, 2019 (ADAMS Accession No. ML19340A019).

NuScale indicated that self-opening of the ECCS main valves is assumed to occur after about 12 hours into certain low probability events when the reactor pressure has reduced to very low levels. NuScale specified that the ECCS main valve spring is designed and manufactured as a safety-related component. NuScale indicated that the function of the ECCS main valve spring is not classified as safety-related, because of the low likelihood of the performance of the self-opening function during a reactor event. NuScale stated that the ECCS main valve spring with its 15 +/-5 pounds per square inch gage (psig) operating range will be included in the design specifications for the ECCS valve system. As part of this audit, the NRC staff reviewed NuScale's document, EQ-B020-2140 (Revision 3), "ASME Design Specification for Emergency Core Cooling System Valves," which requires the main valve spring to have an operating requirement of 15 pounds per square inch differential (psid). As indicated in the ECCS valve system audit report dated December 19, 2019, NuScale has a process underway to update EQ-B020-2140 and other design documentation for the ECCS valve system to incorporate lessons

learned from the design demonstration testing conducted to satisfy 10 CFR 50.43(e). The NRC staff considers the current revision of the design specification for the ECCS valves to be acceptable for the NuScale DCA review. With the operating parameters required in the design specifications, the NRC staff considers the ECCS main valve spring to be a component that is important to safety for the NuScale SMR, in accordance with 10 CFR Part 50, Appendix A.

NuScale stated that the design demonstration of the ECCS main valve spring was evaluated as part of the Proof-of-Concept program with pressurized ambient water testing conducted in 2015 by Curtiss-Wright Target Rock. Upon request, NuScale made available NuScale Engineering Report ER-B020-3817 (Revision B, July 26, 2018), "Proof of Concept Test Report for Emergency Core Cooling (ECC) System Valves," which forwards the Target Rock test report, for NRC staff review in the NuScale eRR. Section 5.2, "Main Valve Opening Test," of the Target Rock report specifies that the purpose of the main valve opening test was to determine the actual differential pressure at which the main valve spring force will overcome the main disc seating force due to pressure. Based on its [[ ]], the report shows an operating pressure range of [[ ]] with no test anomalies noted. The report concludes that the test results prove the theory that the spring will open the main disc at low differential pressure. Based on its review, the NRC staff determined that NuScale has performed adequate testing of the ECCS main valve spring to satisfy the NRC requirements in 10 CFR 50.43(e) for demonstration of this design feature of the first-of-a-kind ECCS valve system for the NuScale SMR.

NuScale DCA Part 2, Tier 2, Section 3.9.6.1, "Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints," includes provisions for the qualification of the ECCS valve system in accordance with American Society of Mechanical Engineers (ASME) QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," as endorsed by RG 1.100 (Revision 3). Under ASME Standard QME-1, the NRC staff considers that the qualification of the ECCS valve system will include all aspects of the ECCS valve design specification, including the assumed performance of the main valve spring.

NuScale DCA Part 2, Tier 2, Section 3.9.6.3.2, "Valve Testing," paragraph (3), "Power-Operated Valve Tests," includes provisions for performance testing of the ECCS valve system as part of the Inservice Testing (IST) Program for the NuScale SMR. In particular, the DCA specifies that the NuScale ECCS valve performance assessment testing contains several attributes, including testing of any ECCS valve not opened during exercise testing during NPM shutdown to demonstrate that the valve will open on low RCS pressure while the trip valve remains energized (closed). NuScale stated that this provision will provide assurance that the performance of the ECCS main valve spring will be demonstrated during the plant shutdown process or as part of individual IST activities. Also, NuScale DCA Part 2, Tier 2, Section 3.9.6.4.3, "Inadvertent Actuation Block Test Frequency Alternate Authorization," describes the applicant's request for performance testing of the ECCS valve system as an alternative to the specific requirements in the ASME "Operation and Maintenance of Nuclear Power Plants," Division 1, OM Code: Section IST (OM Code) as incorporated by reference in 10 CFR 50.55a. This alternative will include removal of the ECCS main valves for testing of their inadvertent actuation block (IAB) valves as part of an initial and subsequent sampling IST testing activity. As part of the safety evaluation (SE) for the NuScale SMR DC, the NRC staff proposes to authorize the request as providing an acceptable level of quality and safety as an alternative to specific ASME OM Code requirements. In conducting the alternative testing activity, the removal of the ECCS main valves will allow an opportunity to verify the acceptable operating condition of the individual ECCS main valve spring.

The NRC staff discussed with NuScale personnel, the timing of the self-opening of the five individual ECCS main valves at low differential pressure by each main valve spring. NuScale indicated that the differential pressure conditions locally at each ECCS main valve might vary upon reaching the self-opening condition for the ECCS main valves. Based on its evaluation of pressure transient performance in the NPM, NuScale stated that the self-opening of one ECCS main valve will promptly reduce the pressure below the self-opening condition for the other four ECCS main valves. Further, NuScale indicated that the slow transient aspects at the pressure associated with self-opening of the ECCS main valves will result in any differences in the timing of the opening of the individual valves to not have a significant impact on the evaluation of the plant event.

Conclusion

The NRC staff has evaluated the mechanical aspects of the boron redistribution issue for the NuScale SMR. The NRC staff conclusions related to these mechanical aspects are as follows:

> With respect to the incorporation of the riser holes, NuScale has adequately assessed the potential FIV effects of the four through holes near the top of the riser. It is highly unlikely that shear layer flow instabilities will form over the holes due to the substantial through flow from the riser region to the SG annulus. The NRC staff believes that the small jets that emanate into the SG annulus will be nearly completely diffused by the downward flow through the SG. In the event that the jets are not diffused, NuScale estimates a jet impact force which is an order of magnitude lower than those already considered in the SG tubing structural design. Also, the frequencies of any jet flow oscillations are well above those of the fundamental SG tubing flexural resonances. Finally, the presence of the holes in the riser will not significantly affect riser structural properties, nor will the stress concentrations lead to alternating stresses approaching the material high cycle fatigue limit.
>
> With respect to the self-opening of the ECCS main valves at low differential pressure, NuScale has provided a reasonable demonstration of the performance of the ECCS main valve spring that satisfies 10 CFR 50.43(e) for this design feature of the ECCS valve system. Further, the operating parameters of the ECCS main valve spring will be included in the design specifications such that the ASME QME-1 qualification of the ECCS valve system will include the performance of the main valve spring. Finally, the IST program for the NuScale SMR includes provisions that will provide reasonable assurance of the performance of the ECCS main valve spring as part of the shutdown process or individual IST activities.

Chapter 4 – Reactor **(NRR/DSS/SNRB)**

The NRC staff audited documents to confirm the potential for uneven boron distribution following passive cooling modes was addressed and identified the need for operational procedures for post-event recovery. The NRC staff notes that ER-000-2486, “Analysis Limits Report,” contains a table describing boron dilution mitigation operational procedural requirements for all events which can lead to the potential to dilute the core, including conditions following extending passive cooling modes when containment or downcomer boron concentrations are unknown.

In addition, the NRC staff audited the FSAR Chapter 4 markups in CP-2071 and CP-2104 and confirmed that Chapter 4 would be updated appropriately. Specifically, the NRC staff ensured

that the FSAR markup contained a sufficient description of the potential for uneven boron distribution such that a future combined license (COL) applicant or holder is aware of the importance of this phenomena when developing operational procedures to recover the module following an event.

Chapter 5 - Reactor Coolant System and Connecting Systems **(NRR/DSS/SNRB)**

The subsequent discussion under TR-0516-49416, "Non-Loss-Of-Coolant Accident Analysis Methodology," is applicable to DCA Part 2, Tier 2, Section 5.4.3, "Decay Heat Removal System." Specifically, the NRC staff audit of ECN-0000-8080, "Assessment of Riser Holes Impact on Non-LOCA Events," showed that riser holes have an insignificant impact on steady-state and transient parameters as well as figures of merit for non-LOCA events. Since DCA Part 2, Tier 2, Section 5.4.3, analyzes generic DHRS cooldowns using NRELAP5, the observation that riser holes have an insignificant impact can reasonably be extended to the DHRS performance analyses in DCA Part 2, Tier 2, Section 5.4.3.

In an informal presentation to the NRC staff, NuScale stated that it performed assessments of riser hole impact on DHRS performance to support DCA Part 2, Tier 2, Section 5.4.3. NuScale observed a minimal impact to the DHRS cooldowns analyzed in DCA Part 2, Tier 2, Section 5.4.3, and concluded that no changes to that section were required. NuScale also presented draft figures comparing the results from DCA Part 2, Tier 2, Figures 5.4-11, 5.4-12, and 5.4-13, to results from calculations that included the riser holes, and the NRC staff observed no significant differences.

In addition, the NRC staff audited the FSAR Chapter 5 markups in CP-2092 and confirmed that Chapter 5 would be updated appropriately.

Chapter 6 - Engineered Safety Features **(NRR/DSS/SNSB)**

This section provides a summary of the part of the audit that was performed by the Nuclear Systems Performance Branch (SNSB) to evaluate the impact of the boron redistribution related to design changes in the NuScale DCA Part 2, Tier 2 Section 6.2.1.1 on Containment Structure and the Containment Response Analysis Methodology (CRAM) technical report (TeR) [Reference 1] that is incorporated by reference. The SNSB audit scope was mainly related to the containment thermal-hydraulic design and containment pressure and temperature response that would result from the limiting containment design basis events (DBEs). The audit also required a close collaboration among the Chapter 6 and LOCA topical report (TR) [Reference 2] reviewers due to several overlapping review areas. NuScale exhibited excellent co-operation and understanding throughout the audit, in meeting the NRC staff's emerging audit needs for the NuScale DCA Part 2, Tier 2 Section 6.2.1.1 and the CRAM TeR. The specific technical details for reaching the safety findings regarding the boron redistribution related design changes as referenced in the NuScale SER Section 6.2.1.1 are provided, as follows.

ADAMS document ML19282C504 provides the details of the NRC staff's two-year long audit of the NuScale containment and ventilation systems that was concluded in Fall 2019. The audit report covered the various NuScale containment design conservatisms and all design changes that NuScale proposed during the course of the DCA review until Phase 4. However, during the public meeting held on March 9, 2020 [Reference 3], NuScale informed the NRC staff about its plans for making additional design changes to the NPM in order to address concerns about the potential boron dilution in the core. NuScale presented a list of relevant sections of the FSAR, technical reports, RAI responses, and technical specifications that were expected to be

impacted due to these design changes [Reference 4]. The list included both FSAR Chapter 6 and Containment Response Analysis Methodology Technical Report TR-0516-49084 for potential changes.

NuScale implemented the following additional design changes: (1) changing the high CNV water level ECCS actuation setpoint range from 264-300 inch to 240-264 inch; (2) addition of new ECCS actuation signals on a low RCS pressure (<800 psia, concurrent with elevated CNV pressure and sufficiently high RCS temperature) and a low differential pressure between the RPV and CNV (approximately 15 psid) to account for the ECCS valves' physical design; (3) addition of four ¾ inch holes in the riser section. The SNSB staff conducted an audit to evaluate the impact of these design changes on DCA Part 2, Tier 2 Section 6.2.1.1 and CRAM TeR [Reference 1]. NuScale provided the description and results of the supplemental sensitivity studies performed to evaluate the impact of the ECCS actuation signal changes and addition of the riser holes on the peak containment pressure and wall temperature results [Reference 5]. Table A-9 in Reference 5, summarizes nine CNV model variants studied by NuScale in the ECCS actuation logic and addition of the riser holes. Reference 5 documents the reevaluation of the limiting containment pressure and temperature cases considering the low RCS pressure ECCS actuation and ECCS valve opening on low differential pressure between the RPV and CNV (approximately 15 psid), as well as the addition of riser holes.

Table B-30 in Reference 5 lists five overall limiting primary side DBEs that were reevaluated for the design changes with respect to peak CNV pressure and temperature. The primary side DBEs included three LOCAs and two AOOs, as follows: (1) discharge line break; (2) injection line break; (3) HPV line break; (4) inadvertent reactor vent valve (RVV) opening; and (5) inadvertent RVV opening. Reference 5 provided the detailed description and table for each DBE for various sensitivity studies conducted for the AC and EDSS power availability, RVV/RRV IAB release pressure, single failure, high CNV level setpoint, and whether riser holes were included in the analysis. The tables also summarized the sensitivity results for the ECCS actuation time and the initiating event (i.e., whether it turned out to be the IAB release, high CNV level setpoint, etc.), and the resulting maximum CNV pressure and maximum wall temperature. The document also provided the sensitivity plots for CNV pressure, RCS pressure, RPV-CNV dp, CNV water level, and RCS hotleg temperature. However, no such updated plots were provided in Reference 5 for the secondary side breaks that are originally documented in the CRAM TeR. Per the NRC staff's request, the applicant provided, and the NRC staff audited an additional document [Reference 6] that presents the key RCS and containment pressure/wall temperature, and water level plots for the bounding cases for the limiting secondary side breaks (SLB, FWLB). Summarily, ECCS does not actuate for SLB and FWLB because these secondary events lead to relatively small accumulation of liquid in the CNV due to rather quick containment isolation. The riser holes do not affect the secondary side mass and energy release, either. The additional information provided in Reference 6 corroborates the statements that NuScale made in Reference 5 for the non-limiting nature of the secondary side CNV events.

As a result of the audit, the NRC staff determined that the boron redistribution design changes had not changed the limiting DBEs, or impacted the calculated peak containment pressure (PCP) and the peak containment wall temperature (PCWT) that remained bounded by the values documented in the DCA Part 2, Tier 2, Section 6.2.1.1. The peak containment pressure remained unchanged at 994 psia for the limiting inadvertent RRV opening PCP, and the peak containment wall temperature remained 526 degrees Fahrenheit (ºF) for the limiting injection line break LOCA. The NRC staff noted that the peak CNV pressure is insensitive to the design changes as the loss of EDSS power, and not the containment water level or the low RCS

pressure signals, remains the initiating ECCS actuation signal for the limiting PCP event. This leads to the same progression of actuation of the RRVs and RVVs due to the respective IAB releases at 1000 psid and 900 psid as before and, thus, the same peak CNV pressure (994 psia). Essentially, due to the loss of EDSS power being the initiating event for the limiting PCP case, the ECCS immediately trips but is held by the IAB, so the new logic has no impact on the PCP. Besides, apparently, four ¾ inch holes are not large enough to significantly affect the mass and energy release into the containment and, hence, the CNV pressure and temperature responses. For the limiting PCWT event, CNV pressure has a small sensitivity to CNV water level setpoint, but PCWT (526 ºF) shows no such sensitivity. NuScale presented a sensitivity study of high CNV level ECCS actuation initiating events under a non-Loss-of-EDSS power condition and found the PCP to be less than 994 psia, with and without riser holes. The high CNV level ECCS actuation setpoint value did not impact the limiting CNV pressure or temperature. NuScale also reevaluated the non-limiting primary and secondary cases with all ECCS actuation signals changes and addition of riser holes and verified them to remain non-limiting. No impact of riser holes' addition was seen on PCP or PCWT.

NuScale updated the DCA Part 2, Tier 2 Chapter 6 to document the design changes [Reference 7: Enclosure 1]. The NRC staff's audit of FSAR Chapter 6 documentation found it lacking on the new design changes. Per the NRC staff's requests, NuScale updated FSAR Chapter 6 [Reference 8] to add Low RCS Pressure (800 psia) to Table 6.3-1 (Emergency Core Cooling System Actuation Values). As agreed by the NRC staff, it also added a footnote for Table 6.3-1 stating that additional information for ECCS actuation valves is provided in Table 7.1-4. The NRC staff verified that Table 7.1-4, "Engineered Safety Feature Actuation System Functions," in FSAR Chapter 7 [Reference 9] had been updated accordingly with the correct High Containment Water Level range (240-264 inch). NuScale also updated the CRAM TeR to document the reevaluation of the containment safety analysis for the modified ECCS actuation logic and the addition of riser holes [Reference 1]. The NRC staff also audited Reference 1; and through several audit teleconferences; agreed with the CRAM updates to document the boron redistribution related design changes.

The audit documents showed that the impact of the boron redistribution changes to the SER Section 6.2.1.1 and containment response analysis methodology was insignificant and the NuScale containment design pressure and wall temperature values of 1050 psia and 550 ºF maintain sufficient margins with respect to the limiting peak containment pressure (994 psia) and the limiting peak containment wall temperature (526 ºF). NuScale showed technical justification that the limiting peak containment pressure and peak containment wall temperature values, as documented in the FSAR Section 6.2.1.1, will be bounding for the proposed boron redistribution design changes. NuScale has also provided the FSAR Chapter 6 and Containment Response Analysis Methodology technical report (TR-0516-49084) updates to reflect the proposed design changes. The audit findings support the updates made by Nuscale in the DCA Part 2, Tier 2 Chapter 6, Revision 5 and CRAM TeR (TR-0516-49084), Revision 3.

References

1. NuScale Power, LLC, TR-0516-49084-P, Revision 3, "Containment Response Analysis Methodology," May 2020 (ADAMS Accession No. ML20141L808).

2. NuScale Power, LLC, TR-0516-49422-P-A "Loss-of-Coolant Accident Evaluation Model," Revision 2, July 2020 (ADAMS Accession No. ML20189A644).

3. NuScale Power, LLC, Submittal of Presentation Materials Entitled "Public Meeting Presentation: Topic - Emergency Core Cooling System (ECCS) Boron Distribution," PM-0320-69218, Revision 0 (ADAMS Accession No. ML20072H333).

4. Electronic Reading Room Reference - ECCS Boron Distribution Expected Changes.031720.pdf, “Current Expected Changes (updated as available).”

5. Electronic Reading Room Reference - ECN-A013-7888.pdf, “ECN-Containment Peak Pressure and Temperature (*2341).”

6. Electronic Reading Room Reference - CNV SLB FWLB Power Avail.pdf, “ECN-A013-7888-Discussion and Approved SLB and FWLB Figures.”

7. NuScale Power, LLC Submittal of Changes (LO-0520-70159) to Final Safety Analysis Report, Section 6.2, “Containment Systems,” Section 6.3, “Emergency Core Cooling System,” and Technical Report TR-0516-49084, “Containment Response Analysis Methodology Technical Report,” May 2020 (ML20141N012).

8. Electronic Reading Room Reference - CP-2076 Part 02 FSAR Chapter 6 05-08-2020.pdf, Final FSAR Chapter 6.

9. Electronic Reading Room Reference - CP-2055 Part 02 FSAR Chapter 7 05-11-2020.pdf, Final FSAR Chapter 7.

<u>Chapter 6</u> - Engineered Safety Features **(NRR/DSS/SNRB)**

GDC 33 Exemption Justification

The NRC staff audited documents supporting the justification for the exemption from GDC 33. In particular, the calculations supporting the justification analyzed breaks that were smaller than the LOCA spectrum. These break sizes are referred to as leakage cases.

ECN-0000-8182 is the primary document that contains the analysis of the leakage cases. The document states that the leakage cases are for breaks that less than 2.5 kg/s. In general, the approach was very similar to the extended DHRS operation cases (as described in the subsequent Chapter 15 discussion). The applicant investigated a total of [[ ]] transients. The initial conditions for the analysis were:

- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]

The applicant only analyzed liquid space breaks (discharge line). The basis for this was that these breaks will lose more inventory to containment and are more likely to miss the containment pressure interlock. From discussions with the applicant, steam space breaks will

depressurize the system faster and, therefore, achieve the low RCS pressure ECCS actuation before the containment pressure interlock is achieved. Also, from discussions with the applicant, the steam space breaks will not lose enough inventory to challenge going below the riser hole elevation, compared to liquid space breaks. The break sizes analyzed for this analysis were (all discharge line breaks):

- [[ ]]

The applicant also assumed that the ECCS valves could open on differential pressure (dP) at [[ ]]. From a conversation with the applicant (and review of other sections of the DCA), the nominal opening dP is 15 psi. Additionally, the applicant modeled the RRVs and RVVs separately, such that they could open in a staggered fashion. Per the document titled "RVV and RRV Passive Opening Time Differences due to Hydrostatic Head," there is a difference of [[ ]] between the RVV and RRV opening, with RVVs opening first. Additionally, the applicant forced one run to fail to open on dP of [[ ]] to observe results if they failed to open.

The methodology for the leakage cases is very similar to that for the extended DHRS operation cases. In particular, the mass flow rate through the riser holes and the condensation rate were calculated using one of the methods established for the extended DHRS operation conditions (range from ~0.5 kg/s to ~1 kg/s for condensation and ~2 kg/s to ~3 kg/s for flow through holes). Additionally, the initial conditions for pool temperature and pressurizer level were the same as those in the extended DHRS operation calculations. At each pool temperature, each pressurizer level was investigated.

Of the [[ ]] cases analyzed, only the cases listed below satisfied an interlock (containment pressure) and did not actuate ECCS on low RCS pressure. However, all of the cases predicted that the ECCS valves open on low dP prior to hole uncovery and before 24 hours.

<u>Cases that satisfied containment pressure interlock with AC Power</u>

[[ ]]
[[ ]]
[[ ]]
[[ ]]
[[ ]]

<u>Cases that satisfied containment pressure interlock without AC Power</u>

[[ ]]
[[ ]]
[[ ]]
[[ ]]

Additionally, the applicant extended the [[ ]] with no offsite power case by artificially lowering the ECCS valves opening on dP (this is identified as the DL-noAC-1-2-24 case). Therefore, the ECCS valves opened at 24 hours due to the timer as a result of losing power to the battery chargers (low ELVS).

ECN-0000-8182 presented plots of system response of the DL-noAC-1-2-24 case since it lost the most inventory of all cases analyzed. The document included plots of the calculated mass

flow rates of the system and boron concentration (in hot region, cold region, and containment), including core critical boron concentration at beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC). From the mass flow rate plot, the NRC staff observed that, at ECCS actuation (24 hour low ELVS timer in this case), there was a large mass flow out of the RRV (~25 kg/s) and mass flow out of the lower plenum (~5 kg/s). The NRC staff observed no inflow into the core.

The plots of boron concentration show that as the riser becomes uncovered, the boron concentration in the hot region (i.e., core and riser) increases, while the boron concentration in the cold region (i.e., DC and lower plenum) decreases. Also, of note, when the ECCS valves open, there is a sharp increase in boron concentration in both the hot and cold region. This can be attributed to flashing. The increase in boron concentration in both regions is an indication that there is little mixing between the two regions; one could expect a reduction in boron concentration in the hot region if liquid from the cold region (which has a lower concentration of about 100 parts per million (ppm) before ECCS actuation) were to mix with it. The MOC case was limiting, with over 100 ppm margin between cold region boron concentration and critical boron concentration.

"Additional Boron Transport Results for Leakage Cases," a document containing talking points from a June 22, 2020, audit call, presents results for cases in which the ECCS valves open prior to 24 hours on low dP. Similar to the DL-noAC-1-2-24 case, these cases show an increase in boron concentration in the hot and cold region after the riser is uncovered. However, at the time of ECCS actuation (which, in this case, actuated on low dP), the cold and hot region boron concentrations converge, indicating that the two volumes are interacting. These cases had more margin to the critical boron concentration compared to the DL-noAC-1-2-24 cases.

Change Packages

The NRC staff audited the FSAR Chapter 6 markups in CP-2076 and CP-2104 and confirmed that Chapter 6 would be updated appropriately.

<u>Chapter 7</u> - Instrumentation and Controls **(NRR/DEX/EICA)**

To address potential boron dilution impacts in specific LOCA event progressions, NuScale revised and posted in the eRR the following Instrumentation and Control related documents to address:

- Lowering the high CNV level ECCS actuation analytical limit range from 264" – 300" to 240" – 264" elevation, and

- Adding a new low RCS pressure ECCS actuation at a nominal value of 800 psia when the CNV pressure is >1.0 psia and the RCS temperature is >475 ºF

NuScale I&C design related documents reviewed in eRR:

- ER-0000-2486, Revision 8, "Safety Analysis Limits Report."

- EE-0000-7939, Revision 0, "Small Break LOCA Boron Dilution Analysis."

- ECN-E000-7934, Revision 0, “D3 Assessment (ER-E000-3530) Update for New ECCS Logic.”

- ECN-E011-7937, Revision 0, “Modify MPS Single Failure Analysis (ER-E011-2069) to address boron dilution issue.”

- ECN-0000-8053, Revision 0, “Digital-Based Common Cause Failure of Wide-Range RCS Pressure Signal Coping Analysis (EC-0000-4937).”

- TR-0316-22048-P, Draft Revision 3, “NuScale Steam Supply System Advanced Sensor Technical Report.”

- TR-0616-49121-P, Draft Revision 3, “NuScale Instrument Setpoint Methodology Technical Report.”

- TR-1116-52011-NP, Draft Revision 4, “Technical Specifications Regulatory Conformance and Development.

- DCA Part 2 Tier 1, Tables 2.5-2 and 2.5-4.

- DCA Part 2 Tier 2, FSAR Sections 3.9.1.1.3, 6.2.1.1.2, 6.2.1.3, 6.2.1.3.3, 6.3, 6.3.5, 7.1.5.1.6, 7.1.5.2.1, 7.1.5.2.2. Tables 3.2-1, 6.2-2, 6.3-1, 7.1-4, 7.1-5, 7.1-6, 7.1-18, and 15.0-7. And Figures 7.1-1f, 7.1-1g, 7.1-1n.

- DCA Part 4 Volume 1, Draft Revision 5, “Generic Technical Specifications” 3.3, Instrumentation, and 5.5.10, “Setpoint Program (SP).”

- DCA Part 4 Volume 2, Draft Revision 5, “Generic Technical Specifications Bases” B 3.3.1, MPS Instrumentation, and B 3.5.

Assessment of the resulting NuScale I&C design change documents posted in the eRR is as follows:

<u>ECCS Actuation on Lower Containment Water Level Signal</u>

To address potential boron dilution impacts in specific LOCA event progression, high CNV water level analytical limit range for ECCS actuation is being lowered to 240” – 264” (elevation), which allows for +/- 12” from the nominal ECCS actuation setpoint of 252” (approximate setpoint, actual setpoint established per the Setpoint Program).

Containment water level sensors provide this ECCS actuation signal. Span of the containment level sensors is estimated at 683.5” from the top of the containment to below the reactor recirculation valves. Calibrated span for these sensors is 100”. Containment elevation of 220” lower limit and 320” upper limit is chosen to maintain a 100” calibrated span that envelops the analytical range of 240” to 264” elevation.

This ECCS actuation signal is automatically bypassed when RCS temperature is below 350 ºF (T-3 interlock) and pressurizer level is above 20 percent (L-2 interlock).

The NRC staff verified that all the affected documents have been updated with the revised CNV water level analytical range and the approximate ECCS actuation setpoint is consistent with the analytical limit and actuation delay time updated in the FSAR Table 15.0-7.

The I&C design change related revisions made to the affected NuScale DC documents provide information needed for the NRC staff to complete the SE.

ECCS Actuation on Low Wide-Range RCS Pressure Signal

Low RCS pressure below 900 psia +/- 100 psia (approximate setpoint, actual setpoint established per the Setpoint Program) actuates ECCS during small loss of coolant events to prevent boron distribution gradients in the RCS. This setpoint is chosen to ensure that actuation occurs before significant accumulation of water with a reduced boron concentration can occur in the different regions of the RCS and the containment. This ensures an unanalyzed reactivity transient will not occur during small loss of coolant events in the containment. This ECCS actuation signal includes interlocks to ensure that the ECCS does not actuate during inappropriate evolutions, such as, startup and controlled shutdowns. The low RCS pressure ECCS actuation signal is bypassed when CNV pressure is below 1.0 psia (P-1 interlock) or when RCS Thot is below 475 ºF (T-6 interlock). These interlocks are designed to distinguish between LOCA and non-LOCA initiators to ensure the low RCS pressure signal is always available during a LOCA event but does not lead to ECCS actuation for a non-LOCA event. A LOCA break within the evaluated size spectrum will quickly increase CNV pressure above 1.0 psia. Additionally, a LOCA event will, relative to non-LOCA, drain the pressurizer resulting in RCS depressurization down to the saturation pressure corresponding to the RCS the upper plenum temperature. An interlock temperature at 475 ºF has some margin to the saturation temperature associated with 800 psia (~518 ºF). Conversely, a non-LOCA event will maintain liquid within the pressurizer for a greater period. DHRS cooling drives RCS upper plenum temperature down, and by the time 800 psia is reached RCS Thot is well below 475 ºF for non-LOCA events.

In support of the ECCS initiation logic changes, safety classification of the wide-range RCS pressure elements is upgraded to safety class A1 and Seismic Category I. Accordingly, all effected documents, including FSAR Table 3.2-1 have been updated appropriately.

Narrow-range (400 ºF to 650 ºF) RCS Thot temperature signal is utilized for T-6 interlock, and narrow-range (0 to 20 psia) CNV pressure signal is utilized for P-1 interlock.

Temperature sensors (RTDs) in the NuScale RPV design are mounted in thermowells to isolate the sensing elements from the process environment. Therefore, pressure and temperature environment for environmental qualification is limited to the conditions for the exterior portion of the RTD located in containment. A total of 12 temperature sensors are used to measure RCS hot temperature. An average of 3 temperature sensors is used as input to a safety function module (SFM) in the corresponding separation group of the MPS (module protection system).

Digital based wide-range RCS pressure signal is credited in the plant safety analysis described in Chapter 15 for mitigating the LOCAs from a spectrum of postulated piping breaks inside the CNV. The limiting case analyzed for LOCA boron dilution is a smaller break in the LOCA break spectrum described in Chapter 15. This case has as ECCS actuation on low RCS pressure. In addition, this analysis conservatively assumes that the highest worth rod assembly is not inserted into the core and neglects negative reactivity insertion from xenon. The D3 coping analysis concluded that a best-estimate case which credits xenon reactivity and all-rods-in

condition would not require an earlier ECCS actuation on low RCS pressure to mitigate a boron dilution scenario for any break size or location in the LOCA break spectrum. ECCS actuation in these cases occurs due to high CNV level or low differential pressure across the ECCS valves. Sufficient diversity exists such that ECCS is available to mitigate these small-break LOCA events.

The I&C design change related revisions made to the affected NuScale DC documents provide information needed for the NRC staff to complete the SE.

Chapter 13 - Conduct of Operations **(NRR/DRO/IOLB)**

NuScale did not make any changes to Chapter 13 as a result of the design changes.

Chapter 15 - Transient and Accident Analyses **(NRR/DSS/SNRB)**

As indicated in Section 1 of this audit summary, NuScale informed the NRC staff in early March 2020, that a series of design changes would be required to address an issue related to boron redistribution in the NPM and the response previously provided for Request for Additional Information (RAI) 8930. As discussed in the Chapter 7 portion of this audit summary, the major changes included: (1) lowering the high CNV level ECCS actuation analytical limit to 240" – 264", (2) adding a new low RCS pressure ECCS actuation at a nominal value of 800 psia when the CNV pressure is >1.0 psia and the RCS temperature is >475 ºF, and (3) addition of four riser holes at the midpoint of SGs. The NRC staff examined NuScale's corrective action reports CR-0220-69077 and CR-0120-68681, which provided the technical bases for the changes made. The final values for the new low-pressure settings were developed over time, and the NRC staff was able to perform sensitivity NRELAP5 calculations to review the impact of the logic changes as they were being developed. The discussion in CR-0220-69077 considered steam space LOCAs to be more limiting than liquid space LOCAs since liquid space breaks fill the CNV faster and reach the ECCS trip on high CNV level earlier, allowing less buildup of condensate. The riser holes were added to address boron redistribution in the DHRS cooling mode, where the RCS level can drop below the riser wall, interrupting the natural circulation and causing a buildup of condensation in the downcomer, resulting from non-LOCA events.

Additionally, an evaluation of the ECCS main valve self-opening feature (discussed in the Chapter 3 portion of this audit summary) was added to the evaluation of Chapter 15 events. The springs do not provide any safety related function, but their existence should be considered in potential outcomes at very low pressures, i.e., reactor system pressure approaching atmospheric conditions. Since these springs actuate on local pressure conditions, the RVVs, located higher up in the NPM, will open first, which the NRC staff believed could provide adverse flow behaviors at the core inlet. This low valve dP condition can be reached in the DHRS cooling mode or for small leakage conditions where the low RCS pressure ECCS logic is bypassed and ac power is available, such that the 24-hour timer is also bypassed, or if RCS pressure drops to atmospheric conditions prior to the 24-hour timer. The NRC staff noted that in the calculations provided in ECN-0000-8182, depending on the RCS pressure, RVVs can open several minutes before the RRVs open. The NRC staff also noted that in ECN-0000-8182, NuScale concludes that when the ECCS valves open by the 24-hour timer, it is likely to result in a net out-surge from the core. The NRC staff's sensitivity analysis of similar cases supported this conclusion, in that early RVV openings likely results in flow stagnation and does not develop an adverse large flow in-surge into the core. For additional information regarding ECN-0000-8182, and the leakage cases, see the Chapter 6 discussion in this document.

In addition to ECN-0000-8182, the NRC staff spent significant effort reviewing the primary supporting documents, EC-0000-7929 (for DHRS cooling modes) and EE-0000-7939 (for LOCA and leakage cases).

To review effects of the addition of riser holes on NPM steady-state operation, the NRC staff reviewed supporting analyses and conducted sensitivity cases over a range of form loss factors and observed that the effect on RCS flow was on the order of [[
]]. Therefore, the riser holes resulted in minimum impact on downstream dependent steady-state parameter results, including RCS temperature and pressure, and secondary side temperature, pressure, and steam mass flow rates.

The NRC staff reviewed the NRELAP5 modeling in Base Model document EC-A010-1782 revisions and noted that the SG tubing in the very top and bottom of SGs was missing from the model. The NRC staff notes that the missing area, which includes mostly the straight portion, in the top could allow the model to under-predict the initiation and the amount of condensation occurring in the downcomer. Since this condensation in the top portion of the SGs occurs before natural circulation is interrupted, it can be assumed to mix well within the RCS fluid. After the riser clears and natural circulation is interrupted, the tube condensate can pool in the downcomer without mixing. The NRC staff questioned NuScale about this discrepancy in several audit meetings, but no quantitative assessments were provided. NuScale indicated that the difference of these upper tubes is small and noted that in EC-0000-7929, NRELAP5 is not used to compute condensation; however, it is used for EE-0000-7939 and ECN-0000-8182. In conclusion of this matter, the NRC staff agreed with NuScale since the straight portion of tubes is relatively short in comparison to the helical section of tubes as the level drops in the downcomer.

The NRC staff reviewed ECN-A030-8055 and confirmed that the form losses for the riser holes, computed using Idelchik, were reasonable for flow through the riser wall to the downcomer.

Effect of Design Changes on Chapter 15 Non-LOCA Event Short-Term Response

The subsequent discussion under TR-0516-49416, “Non-Loss-Of-Coolant Accident Analysis Methodology,” is also applicable to non-LOCA events analyzed in Chapter 15. Specifically, the NRC staff audit of ECN-0000-8080 showed that riser holes have an insignificant impact on steady-state and transient parameters as well as figures of merit for non-LOCA events.

The NRC staff also reviewed materials outside the scope of this audit (e.g., transient progressions in the NuScale DCA) and performed confirmatory analyses using NRELAP5 to help determine whether the ECCS actuation signal changes could result in unnecessary ECCS actuations for non-LOCA events and determined that such scenarios are unlikely based on the interlocks described in further detail under Chapter 7 of this audit summary.

Long-Term Decay Heat Removal System Operation

As part of the design changes, the applicant added riser holes to transport boron mass from the riser to downcomer. The purpose of the riser holes is to minimize downcomer dilution during uncovered DHRS operation, so a diluted slug of water does not cause a reactivity excursion upon opening the ECCS valves at either the 24-hour timer or when the main valve low differential pressure threshold is reached. The NRC staff audited the applicant’s analysis package EC-0000-7929, “Evaluation of Boron Transport during Extended DHRS Operation,” which evaluates the riser hole design to limit the downcomer boron dilution.

Riser flow rate mass transfer was evaluated by the applicant depending on the predominant heat transfer mode. When the primary side level is above the secondary side level, the primary heat transfer mode is convection. When the primary level is below secondary side level, for example during a LOCA, the riser hole mass flow rate is determined by the swell caused by the core/riser void fraction.

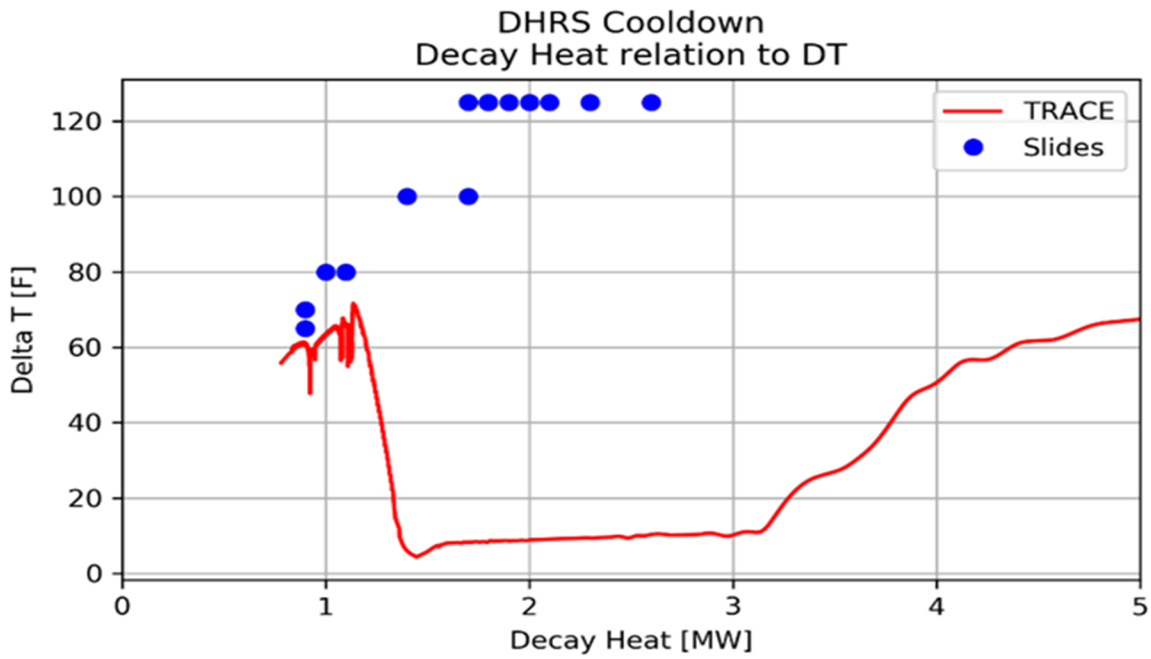
Convective Heat Transfer Mode

For the convective heat transfer mode, the NRC staff reviewed the applicant's calculation of the RCS condensation rate, through wall heat transfer, and riser-to-downcomer level difference at the time of uncovery. The NRC staff found the applicant's calculation of condensation to be conservative, as the DHRS ability to remove heat (i.e., mass flow and temperature difference) was maximized. The DHRS heat removal capability was then converted into an RCS condensation rate independent of the exposed SG surface area. The NRC staff compared the applicant's hand-calculated condensation rate with the applicant's NRELAP5 values which were presented in Appendix F of EC-0000-7929 (see figure below). The NRC staff also compared the applicant's hand-calculated valves to the NRC staff's TRACE confirmatory code as shown below.
[[

]]

Based on the bounding assumptions used by the applicant's hand calculation and comparison to NRELAP5 and TRACE condensation rates, the NRC staff observed the applicant's values are conservative.

To determine the riser hole mass flow rate, the applicant determined the riser and downcomer level difference [[ ]]. To perform this calculation, the applicant assumes a riser-to-downcomer temperature difference as a function of time. This temperature difference [[
]]. A direct comparison between the assumed hand calculation riser and downcomer temperature difference and NRELAP5 or TRACE codes is not possible shortly after riser uncovery, as both codes show liquid discharge over the riser wall. A comparison after liquid discharge between the applicant's assumed values (labeled as "Slides" below) and TRACE is given below.



Liquid discharge over the riser wall stops in TRACE around 1.1 MW. The NRC staff observed that the applicant values are reasonable after 1.1 MW (about 7.5 hours after trip). The applicant conservatively assumes no boron transport during the liquid discharge phase.

The differential pressure at the riser holes, based on the level differences, is converted into mass hole flow rate using a form loss factor of [[ ]]. The applicant estimated a nominal form loss of [[ ]]. The NRC staff reviewed a number of possible form loss factors, such as a thin wall with orifice and the NRELAP5 full abrupt area change (k=2.6) form loss model, and observes that a [[ ]] form loss factor for the convective mode is conservative.

After the first uncovered state point, the applicant's hand calculation assumes the riser temperature decreases from [[ ]] while assuming the same [[ ]] temperature riser-to-downcomer difference. This temperature reduction reduces the riser-to-downcomer density difference by approximately [[ ]]. The applicant stated that this is a conservatism in the hand calculated flow rate. The NRC staff found that the application of this assumption better matched the TRACE confirmatory results.
[[

]]

The NRC staff also compared the applicant's [[ ]], hole mass flow rate, documented in Appendix F of EC-0000-7929, with the hand calculation values and notes the hand calculated flows to be less than the NRELAP5 values as shown below

Time (hrs)	NRELAP5 hole mass flow rate ([[ ]], 52 percent PZR level) (kg/s)	Hand calculation hole mass flow rate (kg/s)
7	[[ ]]	[[ ]]
17	[[ ]]	[[ ]]
24	[[ ]]	[[ ]]

As part of the hand-calculated hole flow rate, the applicant performs an energy balance verification to determine if the assumed riser and downcomer temperature difference is conservative. There are three parts to the energy balance: 1) condensation on the SG, 2) through wall heat transfer, and 3) riser hole flow. These three energy terms are compared to the total decay heat to demonstrate a conservative hole flow rate assuming the other two terms have been conservatively determined. As described above, the NRC staff concluded that the condensation energy term is conservative. The NRC staff also reviewed the applicant's through wall heat transfer calculation, which uses a [[ ]] heat transfer correlation for natural convection flow to calculate the convective heat transfer coefficient. The applicant determined a range of bulk fluid temperatures and a corresponding range of riser wall minus bulk temperatures. From the range of bulk and riser wall temperatures, the applicant determined a heat transfer coefficient and applied it to the inner and outer riser wall surface for each axial zone, the lower riser, transition, and upper riser sections. While separate heat transfer resistances are calculated for the inside and outside riser surface areas, only one heat transfer coefficient is assumed, simplifying the riser heat transfer determination. The NRC staff reviewed the NRELAP5 bottom, middle, and top wall and bulk temperature differences on both the riser and downcomer side to understand the applicant's choice of the convective heat transfer coefficient. The NRC staff reviewed the applicant's choice of riser wall thermal conductivity for an assumed riser wall temperature of [[ ]]. The NRC staff notes the wall heat transfer resistance is dominated by the riser wall conduction by a factor of 4 to 7 times, depending on riser wall thickness [[ ]], compared to the convective heat transfer. Based on the NRC staff's review of the convection and conduction heat transfer resistance calculation, the NRC staff agrees that the through wall heat transfer was conservative. The applicant shows, and staff agrees, that the sum of the individual energy terms is less than the decay heat assumed. The energy balance shows margin to the decay heat value of [[ ]] early in the transient to about [[ ]] at 72 hours. The NRC staff agrees that condensation and through riser wall energy terms are conservative (maximized) and agrees the applicant's energy balance demonstrates a conservative riser hole flow rate within the simplified assumptions of the hand calculation. As described above, the NRC staff noted that assuming the riser was [[ ]] at the second time point was a key element in determining a reasonable riser hole flow rate out to 72 hours.

In EC-0000-7929, the applicant also demonstrated the riser hole placement was adequate to maintain RCS liquid inventory above the riser hole assuming a non-LOCA event. In the determination of the riser hole axial location, the applicant assumed an RCS mass inventory assuming hot full power conditions at a 20 percent pressurizer level consistent with the low-low pressurizer level containment isolation signal as given in DCA Part 2, Tier 2, Chapter 7, Table

7.1-4, “Engineered Safety Feature Actuation System Functions.” The 20 percent pressurizer level containment isolation signal was used to bound non-LOCA loss of inventory events, such as SG tube failure and breaks outside of containment. The applicant assumed an RCS mass of [[ ]], while the NRC staff estimated a minimum inventory of around 46,000 kg. With the minimum RCS mass determined, the applicant converted this mass to a liquid volume using a conservatively high liquid density of [[ ]]. With the total liquid volume determined, the applicant determined the liquid distribution between hot (riser, core and pressurizer) and cold (downcomer and core barrel) sides, accounting for the lower plenum volume, assuming equal hot and cold side levels. The NRC staff review of NRELAP5 results show a 1.5 to 2.0 ft riser-to-downcomer elevation difference after riser uncovery. Therefore, the applicant’s assumption of equal levels is conservative. Considering the conservative liquid level calculation and the riser holes placed 0.6 meters below this level, the NRC staff agrees that the riser holes remain covered by liquid water during non-LOCA cooldowns.

Boiling/Condensing Heat Transfer Mode

The applicant also analyzed postulated scenarios involving extended DHRS operation with the secondary side water level in the SG tubes above the primary side water level in the reactor vessel downcomer, hereafter referred to as the boiling/condensing heat transfer mode. In this scenario, the boiling of borated primary coolant in the reactor core generates steam, which subsequently condenses on the section of the SG tubes that is filled with secondary coolant and above the downcomer water level. The resulting formation of condensate, if left unmitigated, would tend to dilute the boron concentration of the coolant in the downcomer.

To demonstrate that the coolant in the downcomer would not experience excessive boron dilution under boiling/condensing heat transfer conditions, the applicant performed an analysis of the mixing flow between the riser and downcomer via the riser flow holes. This analysis was primarily contained in calculation EC-0000-7929.

Flow through the riser flow holes is governed by the local pressure difference at the flow holes; in this scenario, the local pressure difference is driven largely by the difference in hydrostatic head above the riser flow holes on each side of the riser wall. In particular, due to boiling in the reactor core, the hydrostatic balance at the riser flow holes depends significantly upon the two-phase mixture level on the hot side of the primary system (i.e., in the core and riser regions).

Out of concern that the NRELAP5 code may overpredict two-phase level swell, and hence the driving head for mixing flow through the riser flow holes, the applicant performed spreadsheet calculations to estimate the two-phase mixture level during extended DHRS operation. Although statepoint information from the NRELAP5 code was used (i.e., RCS pressure and decay heat), actual calculation of the two-phase mixture level was performed using the [[ ]].[1] The applicant described its methodology for performing a two-phase mixture level calculation using the [[ ]] primarily in Section 3.4.2 of calculation EC-0000-7929.

As a confirmatory check, the applicant further performed a second mixture level calculation using the [[ ]] for bubbly flow under pool boiling conditions that is described further in [[

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[1] [[ ]]

]]." The applicant stated that its predictions of two-phase mixture level using the [[ ]] were conservatively lower than those calculated by the [[ ]]. The applicant described its methodology for performing a two-phase mixture level calculation using the [[ ]] primarily in Appendix E of calculation EC-0000-7929.

The applicant summarized the mixture level results it calculated using the [[ ]] in Table 4.4-4 of calculation EC-0000-7929. The calculated results in Table 4.4-4 pertain to the scenario with no reactor coolant system leakage, which was the focus of the NRC staff's audit; the applicant ultimately did not credit the two-phase mixture level methodology for scenarios involving leakage. Although there is no mass loss in this scenario, heat removal cools down the RCS, which causes its fluid inventory to shrink. At the final analyzed statepoint, at 72 hours into the event, the applicant calculated a two-phase mixture level in the riser of approximately 13.4 meters. This two-phase mixture level would cover the riser flow holes by approximately 2.4 meters and result in a driving head at the flow holes of approximately 0.4 pounds per square inch.

As described in Section 4.4.4 of calculation EC-0000-7929, the applicant calculated that the mixing flow generated by the two-phase mixture level in the riser for the boiling/condensing heat transfer mode scenario would be sufficient to satisfy the acceptance criteria for downcomer boron concentration, namely, exceeding the critical boron concentration for BOC and MOC conditions, and negligible boron dilution for EOC conditions.

The applicant identified in calculation EC-0000-7929 several conservatisms associated with its calculation of the two-phase mixture level, including the following:

- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]

As discussed above, the applicant's methodology for the boiling/condensing heat transfer mode is similar to the convective heat transfer mode with two notable differences: 1) [[ ]], and 2) the flow through the riser holes is primarily driven by the hydrostatic head above the riser flow holes on

each side of the riser wall (i.e., the two-phase mixture level in the core/riser region and downcomer). The NRC staff's audit of the methodology used to analyze the boiling/condensing heat transfer mode focused on these two primary differences.

The NRC staff's audit of the applicant's treatment of conditions involving boiling/condensing heat transfer focused on the calculation of the two-phase mixture level in the riser region. Although significant advances have been made over past decades, analytical prediction of mixture level behavior, like many two-phase flow phenomena more generally, remains subject to empirically based limitations and uncertainties. Further, as discussed above, the mixture level prediction in the riser region plays a key role in driving the mixing flow between the riser and downcomer for this scenario. Therefore, the NRC staff paid particular attention to assessing whether the applicant's calculation of riser mixture level in the boiling/condensing heat transfer mode contains adequate conservatism.

As described below, the NRC staff's audit focused on the following general areas:

- Applicability of the drift flux modeling approach described by the applicant in calculation EC-0000-7929, including selected correlations and input parameters (e.g., bubble diameter), to the conditions present during extended DHRS operation for the NuScale reactor design.

- Validation of the drift velocity correlations and calculational methods at conditions representative of extended DHRS operation for the NuScale reactor design.
- Confirmatory calculations performed by the NRC staff, including independent analyses applying additional drift velocity correlations.

- Assessment of the adequacy of the conservative margins associated with the applicant's calculated results.

Since first being proposed by [[ ]] in 1965, drift flux correlations have been used in a wide variety of applications, owing to their simplicity, as well as their robust performance within empirically based constraints. Among these successful applications may be counted the prediction of two-phase mixture level in reactor safety analyses. However, in light of their empirical nature, a given drift flux correlation tends to provide reliable predictions only under specific ranges of geometric configurations and thermal-hydraulic conditions. Thus, careful attention is necessary when selecting among the numerous and varied drift flux correlations and input parameters to assure appropriateness for the intended application domain.

The applicant's analysis of downcomer boron concentration in the boiling/condensing heat transfer mode relies upon both the [[ ]] to predict the two-phase mixture level in the riser region. In both cases, the applicant applied [[ ]].

Regarding the [[ ]], in the NRC staff's experience, this correlation has the potential to overpredict void fraction under some conditions relevant to reactor analysis. In particular, the applicant's predictions of void fraction using the [[ ]]. During the audit, the NRC staff observed that, at reduced pressures characteristic of extended DHRS operation, the predictions of the [[ ]] could be significantly affected by small changes in the assumed bubble diameter.

Regarding the [[ ]], this correlation was developed especially for pool boiling conditions, which may be considered relevant to the boiling/condensing scenario analyzed by the applicant, particularly following the uncovery of the top of the riser. [[ ]] was intended to address a wide range of conditions involving low liquid-phase superficial velocities and large-diameter vessels, for which previous correlations may not provide reliable predictions.

The NRC staff's audit particularly focused upon assessing the applicability of the [[ ]] to extended DHRS operation for the NuScale reactor design in light of the unique reactor design and geometry, including a tall riser section, as well as the use of the correlation over a reduced pressure range.

Ensuring the applicability of empirical correlations is generally best accomplished through validation of the correlations, as implemented using the applicant's calculational procedures, against representative test data. However, the NRC staff's audit review found no evidence that the applicant had performed a validation of the [[ ]] at representative conditions using its intended calculational procedure. In audit discussion, the applicant pointed out that [[ ]] contains comparisons of the [[ ]] correlation against various experimental data. However, the applicant did not address the relevance of key test parameters and resulting data in [[ ]] to the NuScale design. In particular, the NRC staff observed that the validation range in [[ ]] does not address subatmospheric conditions.

In light of these limitations associated with the validation of the drift flux correlations proposed by the applicant, the NRC staff's audit focused significantly upon performing confirmatory calculations to assess the adequacy of the applicant's modeling of the two-phase mixture level during extended DHRS operation. The confirmatory calculations performed by the NRC staff included calculations intended to replicate the applicant's calculations with the [[ ]] models, as well independent calculations using additional drift flux correlations.

The NRC staff performed both sets of confirmatory calculations using an independently developed spreadsheet method similar to that generated by the applicant. The spreadsheet calculation was used to automate the iterative calculations, at each analyzed statepoint, for the [[ ]], the fluid properties, and the drift flux correlations.

The first set of confirmatory calculations using the [[ ]] drift flux models was intended to replicate the applicant's results and further extend them by considering sensitivities to alternative input values and modeling practices. In the pressure range of approximately 120-500 psia, which corresponds to extended DHRS operation scenarios with no RCS leakage, the NRC staff's confirmatory calculations with the [[ ]] reasonably reproduced the results of the applicant. However, at reduced pressures down through the subatmospheric range, which may be characteristic of extended DHRS operation with rRCS leakage, increasing disagreement was observed between the NRC staff's and applicant's calculated results. Based upon audit discussions with the applicant, the NRC staff ultimately attributed much of the disagreement to two refinements applied in the NRC staff's confirmatory calculation to more accurately estimate the hydrostatic head: (1) finer nodalization in the riser volume (i.e., 5 cells instead of [[ ]]) and (2) use of a cell-averaged hydrostatic head term in lieu of [[ ]]. Ultimately, the applicant did not credit its two-phase mixture level

methodology for the scenario with reactor coolant system leakage, which obviated the need to pursue the discrepancy further.

The second set of confirmatory calculations brought in additional drift flux correlations, independent of those proposed by the applicant. The NRC staff performed these independent confirmatory calculations to probe the applicant's analysis more deeply in light of limitations identified during the audit, especially the limited validation effort. The two correlations selected for the independent calculations were the GE-Ramp correlation and the FLASH-6 correlation.[2] These correlations were chosen based upon a number of factors, including their difference in form relative to the correlations chosen by the applicant, their comparison against additional validation data, and their history of successful application to mixture level calculations for reactor safety analysis. The NRC staff's independent confirmatory calculations with these two correlations focused upon the pressure range of 120-500 psia, which is associated with extended DHRS operation without RCS leakage. The results of the independent confirmatory calculations demonstrated good (FLASH-6 correlation) to slightly conservative agreement (GE-Ramp) to the void fractions and two-phase mixture level calculated by the applicant for the assessed range of conditions for the no-leakage scenario.

Finally, the NRC staff's audit assessed the conservative margins associated with the applicant's calculation of two-phase mixture level against inherent modeling uncertainties. In the discussion above concerning the evaluation model for the boiling/condensing heat transfer mode, the NRC staff outlined the conservatisms identified by the applicant, which were generally intended to increase the condensation rate in the downcomer and reduce the calculated mixture level. While the NRC staff recognizes that some of the applicant's conservative modeling practices may have countervailing individual impacts (e.g., [[ ]] tends to increase the predicted mixture level), when taken as a whole, the NRC staff observed that the set of conservatisms identified by the applicant would provide additional margin, albeit unquantified, in the calculated mixture level. This additional conservatism applies particularly to extended DHRS operation scenarios with no RCS leakage, for which the calculated riser level predicted by the applicant would remain above the riser flow holes even with no voiding in the riser.

In the NRC staff's judgment, incorporation of adequate conservative margin in the two-phase mixture level calculation is essential to offset inherent calculational uncertainties. For instance, as discussed above, the applicant's calculations did not include validation of its proposed [[ ]]. Furthermore, various effects such as bubble agglomeration and wake effects, which could lead to larger drift velocities and reduced void fractions along the 11-meter height of the riser, were not explicitly addressed in the mixture level calculation audited by the NRC staff. In light of the conservatisms in the applicant's calculations, as well as the relatively low riser void fractions calculated for the extended DHRS operation scenario with

[2] For the GE-Ramp correlation, see Manera, Annalisa & Prasser, Horst-Michael & Hagen, Tim (2005) "Suitability of Drift-Flux Models, Void-Fraction Evolution, and 3-D Flow Pattern Visualization During Stationary and Transient Flashing Flow in a Vertical Pipe," *Nuclear Technology*, 152: 38-53. For the FLASH-6 correlation, see WAPD-TM-1249, "FLASH6: A FORTRAN IV Computer Program for Reactor Plant Loss-of-Coolant Accident Analysis (LWBR Development Program)," July 1976.

no RCS leakage (i.e., ≤ 7 percent) that reduces the potential for interactions between bubbles, the NRC staff observed that these uncertainties appear unlikely to have a major impact.[3]

Based upon the information reviewed and confirmatory calculations performed during the audit, the NRC staff observed that the mixture level in the riser would likely exceed the value calculated by the applicant during extended DHRS operation scenarios involving no RCS leakage. This observation is based primarily upon the following points that have been elaborated further in the discussion above:

- The applicant's methodology, apparently in part due to its parameterization of the bubble size used in the [[ ]], calculated void fractions in the riser that were less than those calculated by the [[ ]], for which validation comparisons exist in [[ ]] that are relevant to extended DHRS operation scenarios with no RCS leakage.

- The NRC staff performed confirmatory calculations, both with the [[ ]] used by the applicant and independent analyses using additional correlations with diverse validation bases. These confirmatory calculations were consistent with or predicted higher mixture levels than those predicted by the applicant for extended DHRS operation scenarios with no RCS leakage.

- As discussed above, the applicant included conservatism in its calculation of the two-phase mixture level that could provide margin to offset calculational uncertainties expected at the low-void conditions anticipated during extended DHRS operation scenarios with no RCS leakage.

Furthermore, while the NRC staff observed that the applicant's results and methods for calculating two-phase mixture level appear reasonable for the specific set of conditions associated with the proposed application (i.e., the scenario involving no RCS leakage), significant limitations concerning the validation of the applicant's proposed methodology were identified during the audit outside of this limited range of conditions. In particular, the NRC staff observed evidence during the audit indicating that the methodology proposed by the applicant may not make reliable predictions at conditions (e.g., pressure, void fraction, decay heat) associated with the scenario involving RCS leakage.

To determine the riser hole mass flow rate, the pressure differential across the hole must be determined, which is largely governed by the hydrostatic head above the riser flow holes on each side of the riser wall. To determine the pressure difference, the applicant performs a [[ ]]. The pressure differential is converted into the hole mass flow rate assuming a conservative loss coefficient. The applicant states the riser hole geometry is such that [[ ]]. For the boiling/condensation heat removal mode, as discussed in Section 2.2.1 of EC-0000-7929, the applicant applied a [[ ]] uncertainty to the nominal form loss. The NRC staff's audit

[3] For instance, concerning the impact of void fraction on wake effects, see Tomio OKAWA , Kimitoshi YONEDA , Shirong ZHOU & Hiroaki TABATA (1999) New Interfacial Drag Force Model Including Effect of Bubble Wake, (II), Journal of Nuclear Science and Technology, 36:11, 1030-1040, DOI: 10.1080/18811248.1999.9726295.

observed that the use of this uncertainty factor adds margin that is intended to address uncertainty associated with the true value of the hole form loss.

Calculation of Downcomer Boron Concentration

The applicant's acceptance criterion is that the time dependent, minimum downcomer boron concentration remains above the critical boron concentration except at EOC conditions. The NRC staff agrees this is a conservative criterion, as it precludes a return to power with significant excess reactivity (i.e., BOC or MOC conditions). The time-dependent boron concentration is calculated using a [[
]]. The use of this model effectively assumes [[
]]. For the convective heat transfer mode, the applicant demonstrated, except for EOC conditions, that the minimum downcomer boron concentration remained above the critical boron concentration assuming the highest worth control rod is stuck out, no core voiding, and no xenon worth down to 200 F. The applicant calculated the critical boron concentration using SIMULATE-5 (approved as described in Section 15.0.2 of the NRC staff's SE) with the associated reactivity uncertainty (nuclear reliability factor). Beginning and middle of cycle values for downcomer boron concentrations remain above the critical value with margins of 246 ppm and 75 ppm at 72 hours, respectively. The margins at the 24-hour timer increase, as the downcomer concentration is approximately the same, but the effects of xenon decay decrease the critical boron concentration. The EOC downcomer and critical boron concentration comparison indicates a potential return to power at EOC. As discussed in Section 15.0.6 of the NRC staff's SE, a return to power during a riser uncovery under nominal conditions is not expected but is possible if reactivity uncertainties are included. .

For the boiling and condensing heat removal mode, the same methodology applies, but for the xenon worth versus time, the applicant assumes an initial hot full power equilibrium xenon concentration. The minimum margin case is at BOC conditions around 2 hours, which is near the time of the first downcomer turnover. At 72 hours, there is 158 ppm margin at BOC and a 32 ppm margin at MOC. The margins at the 24-hour timer increase, as the downcomer concentration is approximately the same, but the effects of Xenon decay decrease the critical boron concentration. Based on the conservative [[
]] and the conservatisms in the condensation and hole flow rate, the NRC staff observed that the downcomer concentration remains above the critical boron concentration, which assumes no core voiding. The minimum time-dependent margin increases at MOC and, as with the convective heat transfer mode, a potential return to power is bounded by the covered riser and ECCS cooldowns as discussed in Section 15.0.6 of the NRC staff's SE.

Review of supporting calculations for RAI 8930 related to Long-Term Cooling Condensation Effects

The NRC staff reviewed ER-0000-6621, EC-0000-7110, EC-0000-7127, and EC-0000-7144, from the Chapter 15 audit materials to re-review the NRELAP5 boundary conditions selected for the various long-term cooling (LTC) cases that supplied input to the boron dilution spreadsheet to determine if maximizing condensation in the containment and the downcomer had been considered as a figure of merit. The NRC staff noted that all the EC-0000-7110 cases used a high pool temperature of 210 °F, and the NRC staff became concerned that lower pool temperatures could result in more condensation and had not been considered. This pool temperature selection was the subject of several audit meeting discussions in August/September of 2019. At the time, the higher pool temperature was considered

reasonable by staff since it would maximize temperature and voiding in the core and maximize boron volatility losses, and lower pool temperatures were not considered. In light of recent concerns about post-ECCS actuation condensation in the downcomer and CNV, and the effects of potential long-term boron dilution at the core inlet, the NRC staff initiated a series of NRELAP5 sensitivity cases using the modified LOCA modeling (with riser holes and new ECCS settings) to review condensation production capability of the limiting RRV inadvertent opening cases used in the disposition of RAI 8930. The NRC staff ran cases with pool temperatures at 60 °F, 100 °F, and 200 °F, and determined that the amount of condensation produced during ECCS cooldown (with or without DHRS) is not very sensitive to the pool temperature. This is because of the large condensation surface areas, such that even at higher pool temperatures, all the steam produced in the core via decay heat cooldown is condensed. At lower pool temperatures, essentially the same amount of condensate (distilled water) is produced. This therefore indicated to staff that the amount of condensation produced in the downcomer and CNV in the limiting boron dilution cases provided for the RAI 8930 response are reasonable.

Effect of Design Changes on Chapter 15 LOCA Event Short-Term Response

The discussion under TR-0516-49422, “Loss-Of-Coolant Accident Analysis Methodology,” reviews the effect of relative changes for the LOCA topical report as applied to DCA Part 2, Tier 2, Section 15.6.5. The NRC staff audited the LOCA break spectrum supporting calculation, EC-0000-2749, which showed that riser holes have a negligible effect on transient results but that the addition of the new low pressure ECCS signal affected most cases in the spectrum, causing significantly earlier ECCS actuation. Since the limiting case for minimum critical heat flux ratio (MCHFR) and minimum core collapsed liquid level do not credit DHRS and did not reach the low pressure setting, the limiting results for these key figures of merit remained unchanged for the LOCA events.

Effect of Design Changes on Chapter 15 LOCA Event Long-Term Response

The NRC staff audited the LTC calculation, ECN-A010-8071, which indicated that riser holes have a negligible effect on transient results. It also indicates that the new low pressure ECCS signal has little impact for long term cooling scenarios. For LOCA events post-ECCS actuation, the riser holes will be uncovered and essentially have no impact long term. Therefore, the limiting cases for minimum and maximum RCS temperatures and other figures of merit remained unchanged for the LTC events.

Change Packages

The NRC staff audited the NuScale change packages for TR-0916-51299-P, “Long-Term Cooling Methodology,” (CP-2066) to confirm that the technical report would be updated appropriately. The NRC staff provided minor comments on the changes proposed as the changes also were minor. The NRC staff wanted to make sure that the new ECCS logic for the low pressure signal and the riser holes were adequately described and their effect evaluated. The NRC staff agreed that based on the evaluations in ECN-A010-8071 and ECN-0000-8080, the design changes have minimal impact for the LTC TR.

In addition, the NRC staff audited the FSAR Chapter 15 markups in CP-2050, CP-2071, and CP-2104 and confirmed that Chapter 15 would be updated appropriately.

<u>Chapter 16</u> - Technical Specifications **(NRR/DSS/STSB)**

The NRC staff reviewed draft markups of DCA Part 2 Chapter 16; Technical Report (TR)-116-52011-NP, Revision 4, "Technical Specifications Regulatory Conformance and Development"; and DCA Part 4, generic technical specifications and bases, and provided the applicant feedback concerning minor editorial clarifications. The NRC staff found that these clarifications were included in the May 20, 2020, DCA Revision 4.1. The NRC staff found that the changes to these parts of the application adequately reflect the design changes to address the boron redistribution issue.

Chapter 19 - Probabilistic Risk Assessment and Severe Accident Evaluation **(NRR/DRA/APLC)**

Audit Conference Calls:

Audit conference calls were held between NuScale and the NRC staff on April 13, 2020; April 30, 2020; May 6, 2020; May 8, 2020; May 15, 2020; May 26, 2020; June 5, 2020; June 12, 2020; and June 16, 2020; to discuss the various aspects of the boron redistribution issues with respect to the NuScale PRA. A summary of issues discussed at these audit calls is provided in Table 1 below. In the following section, Table 2, the list of audited documents includes NuScale's responses to the NRC staff's questions during the audit calls.

Table 1 - Audit Call Discussion Summary

Date of Audit Call	Topics	Discussion Summary
April 13	Scope of PRA impact assessment	The applicant discussed the proposed contents of two reports to document the PRA related impact assessment of the design changes. The NRC staff provided initial feedback on information needed to reach a regulatory finding.
April 30	CVCS line break outside containment LOCA inside CNV with CNV isolation failure	The applicant and the NRC staff discussed ER-P060-8072 and ER-P000-8076 documenting the PRA related assessment of the design. Specific discussion topics included: 1) Unisolated CVCS injection line break outside containment • Extent of DHRS heat removal credited after SGs are uncovered • Adequacy of DHRS model validation for conditions where SGs are uncovered • Applicability of DHRS passive safety system reliability analysis to LOCA scenarios • Impact of riser holes on DHRS passive safety system reliability analysis • Dependency between ECCS and operator action to initiate CFDS (operator cues; how task performed) 2) LOCAs inside containment with containment isolation failure (e.g., inadvertent opening of an RVV with CES line isolation failure) with respect to the possibility of earlier ECCS actuation resulting in higher loss of coolant to outside CNV 3) Impact of operation with containment pressure greater than 1 psia but within containment leakage TS (~ 2 psia) on ECCS actuation frequency

		The NRC staff discussed the following additional information needs: 1) Plots for unisolated CVCS injection line break outside containment scenario 2) Sensitivity study with DHRS heat removal turned off after downcomer level drops below bottom of SGs; plots as listed above 3) Plots for inadvertent opening of an RVV with CES line isolation failure scenario
May 6	LOCA break spectrum and locations considered in PRA	The NRC staff noted in Chapter 15 audit information (e.g., EE-0000-7939) that breaks on the small end of the spectrum that result in delayed ECCS actuation is generally more limiting for boron dilution due to more potential condensation and colder RCS conditions. PRA LOCA success criteria analyses assume a double ended break which minimizes the time to ECCS actuation. The NRC staff requested the applicant to discuss how EE-0000-7939 information was evaluated for impact on PRA inside containment LOCA-ATWS scenarios or other assessments of boron redistribution impact on PRA LOCA success criteria. Especially since the PRA LOCA analyses assumed a double ended break which minimizes the time to ECCS actuation and partial breaks are probably more likely and more consequential in this case. The applicant agreed to provide N-RELAP analysis addressing the NRC staff concern.
May 8	LOCA-ATWS	The applicant and the NRC staff discussed the additional N-RELAP5 SLOCA-ATWS thermal-hydraulic analyses considering the break spectrum to evaluate the ECCS design change effectiveness to limit boron redistribution. The N-RELAP runs show that the ECCS actuates before significant downcomer dilution would occur.
May 15	Operator actions for small leaks FSAR key assumptions on break sizes Effect of operation with higher initial CNV pressure on ECCS actuations LOCA ATWS	The applicant and staff discussed FSAR Chapter 19 revisions to include design features (ECCS actuation and riser holes) and key assumptions relevant to the boron redistribution issues. The applicant and the NRC staff also discussed the following audit questions: 1) GDC 33 exemption information indicates that small leaks (< ¼ inch) are mitigated by procedures and operator actions. How is this modeled in the PRA? 2) Smaller breaks seem to be most challenging with respect to boron redistribution before ECCS actuation. Guillotine breaks are identified as bounding in the key assumptions for the full power success criteria in DCA Table 19.1-21. The success criteria do not discuss riser uncovery phenomena which the NRC staff believes is most limiting for recriticality concerns. The NRC staff requests that NuScale discuss very small breaks and riser uncovery in this key assumption table under success criteria. 3) LCO 3.5.3, UHS, requires Reactor Pool temperature at or below 110 degrees F and greater than or equal to 65 degrees. Based

	ECCS trip information FSAR key assumptions on riser holes	on the CNV pressure versus UHS pool temperature plot on FSAR page 5.2-47, operation with CNV pressures up to 1 psia can occur. Isolated injection line LOCAs outside of containment with RCS leakage is evaluated in Section 5.2.1.4 of the PRA T/H analysis. How does elevated initial CNV pressure impact the T/H analysis of isolated LOCAs outside containment if initial CNV pressure was roughly 1 psia? Would hot leg temperature remain above 475F when RCS pressure reaches [[ ]] psia? The concern is that this event could lead to ECCS actuation. 4) The NRC staff would like confirmation that LOCAs with ATWS cases provided to the NRC staff actuated ECCS on low RCS pressure setpoint versus CNV level where the potential for additional dilution is increased. Could NuScale provide the CNV level plots for these cases? 5) eRR report 8076, Section 3.2, "Description of the Riser Holes," states that downcomer dilution scenarios are more likely to occur during non LOCA scenarios. Riser bypass holes would eliminate the concern of downcomer deboration and support indefinite DHRS operation. Calculation 7939 also indicates that downcomer deboration could be a concern during small LOCAs.
May 26	N-RELAP LOCA ATWS Modeling Uncertainty	The NRC staff requested an audit call with NuScale to understand what evaluations and assessments that may have been performed to gain confidence in the N-RELAP's boron tracking model capability to support the PRA ATWS T-H analyses.
June 5	Post-ECCS CFDS Injection LOCA ATWS Modeling Uncertainty Kinetics Parameters supporting the N-RELAP ATWS analysis	The applicant and the NRC staff discussed the reactivity consequence of post-ECCS CFDS injection. The applicant estimated the core flow rate following 100 gpm injection of CFDS water into the CNV (.05 inches/sec) . The applicant stated that there will be mixing between the diluted downcomer water and lower plenum water. The applicant concluded that core damage would not occur under this scenario. The NRC staff raised the following questions regarding the adequacy of uncertainty assessments performed for the N-RELAP5 LOCA-ATWS analysis 1) Please describe how the PRA NRELAP5 boron methodology was validated/verified for this analysis to gain confidence in the results supporting Chapter 19. 2) Was any benchmarking performed (e.g., against Chapter 15 boron transport methodology)? 3) What are the key reactivity components in preventing core damage and what is their relative importance? 4) What are the key sources of uncertainty in the analysis? 5) Can NuScale provide a qualitative evaluation to augment the NRELAP5 analysis? (e.g., qualitative evaluation for CFDS injection following long term ECCS cooling)? 6) Can NuScale's conclusion on LOCA ATWS not resulting in core damage can be augmented with sensitivity calculations?

		The NRC staff requested information on how kinetics parameters pertaining to boron and moderator reactivity are calculated and input to the model.
June 12	N-RELAP Sensitivity Studies Hand Calculation Using the Fuchs-Nordheim Model	• The applicant discussed additional evaluations performed to address the uncertainties in the LOCA ATWS analysis. The applicant presented a sensitivity analysis considering a later ECCS actuation simulated by a lower RCS pressure setpoint, which increases the dilution in the downcomer. The applicant calculated the energy deposition in the fuel of [[ ]] for these scenarios showing substantial margin to the core coolability criteria of 230 cal/g per SRP 4.2. The applicant supplemented the code calculation with a hand calculation using the Fuchs Nordheim method reached consistent conclusions.
June 16	Non-LOCA ATWS	The NRC staff requested the applicant to confirm the NRC staff understanding below regarding the non-LOCA ATWS scenario: 1) for the non-limiting ATWS case where no control rods insert, the RCS pressure will remain above [[ ]] psia indefinitely. 2) the PRA considers any event where with the failure of 3 out of 16 control rods to insert as an ATWS. 3) the operators would be expected to manually trip the CRDMS or manually deenergize the CRDMs. 4) In an ATWS event, the RSVs are assumed to be demanded. RSV cycling is assumed to lead to ECCS actuation. The applicant stated that the NRC staff's understanding above is accurate. The NRC staff requested the applicant to address the following questions: 1) Does the event tree assume that the reactor is subcritical before ECCS is actuated? 2) If all the rods fail to insert due to mechanical binding and therefore, operator recovery is not possible though manually tripping the reactor or de-energizing the CRDMs, is CVCS injection effective or even exacerbate the consequences due to void collapse? 3) Does the event tree model common cause failure of all the control rods failing to insert?

Table 2 below lists the audited documents to support the regulatory findings for the NuScale PRA related to boron redistribution. This list of audit documents includes NuScale's responses to staff's questions discussed during the audit calls.

Table 2 – Audited Documents

Document	Summary of Document
ER-P060-8072, "PRA Thermal-Hydraulic Impact Analysis of Boron Redistribution Design Changes"	• Evaluations include changes to ECCS actuation logic and the addition of bypass holes through the upper riser. The (deterministic) small break LOCA analysis boron dilution analysis, EE-0000-7939, concludes that the low RCS pressure signal with hot leg temperature and CNV pressure conditional interlocks will actuate ECCS prior to significant boron redistribution for a very wide range of LOCA conditions. Because of the interlock requirements, the non LOCA scenarios that result in ECCS actuation on the new signal are those with continuous RSV cycling (e.g. DHRS failure) and certain non LOCA ATWS scenarios. • <u>CNV level setpoint of 252 inches above the reactor pool floor is assumed. [changed from 264-300 to 240-264]. Low RCS pressure setpoint of [[ ]] psia assumed.</u> Conditional interlock, hot leg temperature must be greater than 475 F, and Conditional interlock CNV pressure must be greater than 1 psia. Both conditional interlocks must be satisfied for the new ECCS actuation to occur. The interlocks exist to minimize unnecessary ECCS actuations, thereby minimizing the impacts on event sequence progression and the PRA risk results. The combination of RCS pressure, hot leg temperature, and CNV pressure differentiates a small break LOCA into containment (for which ECCS is desired) from a DHRS cooldown with intact RCS boundary (for which ECCS is not desired). • <u>For a DHRS cooldown with intact RCS boundary, the hot leg temperature will be below 475 F when RCS pressure reaches the low RCS pressure setpoint (RCS is subcooled), thereby preventing ECCS actuation.</u> For scenarios involving a loss of RCS inventory, hot leg temperature will be above 475 F when the RCS pressure reaches the low RCS pressure setpoint (RCS is saturated or nearly saturated), thereby enabling ECCS actuation. • Containment pressure interlock is needed because there is no level measurement at the bottom of the CNV. Narrow range CNV pressure sensor is the [[ ]] instrument for identifying a small break into the containment. • Small break LOCA boron dilution analysis shows the ECCS logic changes preclude boron redistribution prior to ECCS actuation. New ECCS actuation setpoints actuate ECCS [[ ]] in the progression of LOCAs in containment. • Reducing the CNV level for ECCS actuation does not introduce new event sequences but it changes the timing

Document	Summary of Document
	of ECCS actuation for those events actuated on a high CNV level. • Riser bypass holes assumed to be placed [[ ]] up the SG, comprised of four ¾ inch diameter holes. Objective is to ensure adequate RCS mixing and preclude any significant boron redistribution during extended DHRS operation. The rate of boron redistribution will be much lower because of reduced condensation on the primary side of the SG. • NRELAP v1.4 used. Base model used in Level 1 PRA success criteria with modifications. Initial CNV pressure is reduced to [[ ]] psia by adjusting the CES boundary condition, time-dependent volume. Reduced CNV level from 282" to 252" relative to pool floor. Added new ECCS actuation logic: all three variable trips (RCS pressure < [[ ]] psia, Thot > 475 F, and CNV pressure > 1 psia) must be "true" for ECCS actuation to occur. Riser bypass holes modelled in the form loss through the junction. • <u>Nominal DHRS cooldown –</u> (TT, LOFW, LOCVCS, LO PZR heater power) result in MPS-reactor trip, containment isolation, and DHRS actuation. Case run with both one and two trains of DHRS available to show new actuation is bypassed because Thot is below 475F when RCS pressure reaches [[ ]] psia. • <u>Single RSV Actuation</u> – 50 percent probability of RSV opening following a non-LOCA transient with successful DHRS. Since RSV lift setpoint of 2075 psia is not reached, the setpoint is reduced by [[ ]] psi to [[ ]] psia. Closing setpoint is reduced to [[ ]] psia to [[ ]] psia. (purpose to show new actuation is bypassed because Thot is below 475F when RCS pressure reaches [[ ]] psia). This case was run with just one train of DHRS. • <u>Secondary line breaks (FWLB, MSLB) inside containment</u> - followed by RT, CI, DHRS. (purpose to show new actuation is bypassed because Thot is below 475F when RCS pressure reaches [[ ]] psia). Since secondary coolant discharges into the containment, CNV pressure is expected to be near the 1.0 psia interlock. • <u>Isolated LOCA outside containment with RCS leakage –</u> CVCS injection line break outside containment, CI, RT, DHRS (on train available). No discharge of coolant into the containment. To confirm that containment pressure will remain below 1 psia, this case models the maximum operating leakage between RCS and CNV following CES isolation to show new actuation is bypassed because CNV pressure is below 1 psia when RCS pressure

Document	Summary of Document
	reaches [[ ]] psia, and CNV pressure remains below 1 psia until Thot is below 475F. • Non LOCA ATWS NRELAP PRA Level 1, PRA success criteria, (ER P060-6892) , for the non-limiting ATWS case where no control rods insert, the RCS pressure will remain above [[ ]] psia indefinitely. The PRA considers any event where with the failure of 3 out of 16 control rods to insert as an ATWS. Additionally, operators would be expected to manually trip the CRDMS or the operator manually deenergizing the CRDMs. The RCS pressure, RCS temperature, and CNV pressure will depend on the number of control rods inserted into the core and any operator actions to manually trip the reactors, or borate the RCS. No additional RELAP runs were executed for this scenario. Non LOCA ATWS are conservatively assumed to result in ECCS actuation and are conservatively represented by LOCA ATWS events because they involve early ECCS demand which is more challenging • <u>RSV Cycling</u> – non-LOCA transient or successfully isolated outside containment LOCA with failure of DHRS will demand RSV cycling until sufficient liquid accumulates in the CNV to cool the RPV outer surface. Once heat removal exceeds core decay heat generation, RSV cycling stops and RCS gradually depressurizes. RCS pressure is predicted to drop below [[ ]] psia within the 72 hour PRA mission time. RCS coolant is saturated, therefore the hot leg temperature will exceed 475 F when RCS pressure drops below [[ ]] psia. Because of the coolant discharge into containment, CNV will exceed 1 psia. Therefore, ECCS will actuate as a result of the new ECCS signal for the RSV cycling cases. • Timing confirmation – impact of earlier ECCS actuation on mitigation actions. LOCAs inside containment that involve successful ECCS are not at risk of changing end state because successful ECCS will keep the core covered and cooled whether it actuates on the current CNV level or any time earlier. • Sequences where ECCS demand timing is most important are those that result in failure of ECCS and are <u>mitigated with coolant makeup via CVCS or where CFDS</u> is used to mitigate unisolated LOCAs outside containment. The end states of these sequences are sensitive to changes in the event progression because the outcome depends on operators restoring coolant before the core uncovers sufficiently for core damage to occur. <u>[[ ]] demand and incomplete ECCS actuation can [[ ]] the loss of coolant and challenge the response time of the coolant makeup action</u>.

Document	Summary of Document
	• LOCA outside containment with CFDS makeup – MPS signals result in demands for RT, CI, DHRS actuation shortly after event initiation. Isolation of CVCS line fails. Operators initiate coolant makeup to the containment via CFDS once they diagnose the scenario. In this case successful ECCS actuation is required for CFDS operation to be successful in preventing core damage. [[ ]] actuation of ECCS has the potential of [[ ]] the loss of coolant from the RPV before the liquid level in containment is sufficiently high to return coolant through the RRVs and into the core. Case run with and without availability of DHRS. • Incomplete ECCS actuation with CVCS makeup – Initiated by CVCS injection line break, or spurious actuation of an RRV. MPS signals result in demands, for RT, CI, and DHRS actuation. ECCS actuates incompletely, with RRVs opening. Operators initiate coolant makeup via CVCS pressurizer spray line once they diagnose the scenario. [[ ]] • DHRS driven cooldown does not result in ECCS actuation on the new signal because both the hot leg temperature interlock and the CNV pressure interlock are not satisfied with the RCS pressure reaches the new setpoint. For single DHRS case, hot leg temperature is approximately [[ ]] F when RCS pressure reaches [[ ]] psia. For the two DHRS case, the hot leg temperature is even lower. • Single RSV may exceed the 1 psia CNV pressure interlock but the temperature is well below 475F interlock when RCS pressure setpoint is reached. • Secondary line breaks (FWLB, MSLB) inside containment - may exceed the 1 psia CNV pressure interlock but the temperature is well below 475F interlock when RCS pressure setpoint is reached. • Isolated LOCA outside containment with RCS leakage – There is an initial loss of RCS inventory. Low RCS pressure is reached with hot leg temperature of about [[ ]] F which satisfies the temperature interlock. However, the CNV pressure remains well below the CNV pressure interlock, even when the maximum leakage rate into CNV is considered. While CNV pressure may eventually exceed the 1.0 psia interlock, the RCS temperature will be below 475 at that time. • LOCA outside containment with CFDS makeup – The new ECCS signal results in ECCS actuation when the level in the CNV is well below the RRV elevation. The result is that following ECCS actuation, the core

Document	Summary of Document
	becomes uncovered, fuel heat up occurs, and core damage occurs before the liquid level can be recovered in the core. With one train of DHRS, RPV pressure is reduced and the rate of coolant loss from the RPV is reduced. Therefore, CFDS is capable of recovering level in the core and preventing core damage. • Letdown line break is less limiting than the charging line break because the elevation of the discharge line is higher than the injection line; therefore, the break flow transitions from liquid to steam earlier and the coolant loss rate is reduced, allowing more time for CFDS to recover level. This case avoids core damage with or without DHRS. • <u>Incomplete ECCS actuation with CVCS makeup –</u> ECCS design changes do not change the end state of these sequences. These liquid break cases result in the most [[ ]] from the RPV to CNV of any event sequence in the PRA. • Successful ECCS with failure of containment isolation (CI) for the Containment Evacuation System – Previous analysis showed that CI is not required for success of ECCS because ECCS operation drives system pressure down to atmospheric conditions with sufficient coolant remaining in the module to keep the core covered. Because the ECCS design change result in earlier ECCS actuation, the system will reach atmospheric pressure earlier and lose less coolant from the module, therefore no impact. The Passive System Reliability Probabilistic Risk Assessments Report Analysis (PSSR) -ER-P10-3777 A [[ ]] CNV level results in [[ ]] ECCS actuation. A [[ ]] level results in [[ ]] ECCS actuation. The assessment referenced in ER-P10-3777 found that for RRV LOCA , [[ ]] with respect to the figure of merit. However, CNV actuation level was found to be significantly less important that than other parameters such as the reactor pool temperature, the RVV flow coefficient, the CNV heat transfer coefficient, That is because the minimum RPV level occurs a considerable amount of time after ECCS actuation and is driven by the physical phenomena that affect the pressure and level imbalance between the RPV and the CNV namely factors that decrease pressure in containment and an increase in pressure in the RPV . The PSSR analysis considered the impact of RCS loop form losses and flow rate on the performance of ECCS and DHRS and found that they had a negligible impact on either system. The riser bypass flow holes design change has the effect of redirecting a small amount of RCS flow though the

Document	Summary of Document
	riser instead of across the entirety of the SG tube bundle. The impact of the riser holes is therefore judged to be similar to that of form losses of performance of the ECCS and DHRS with respect to the PSSR analysis (a negligible impact). This assessment is confirmed in Appendix E of Long Term Cooling Analysis EC-A010-4270, where the riser holes are demonstrated to have no impact on DHRS cooldown characteristics and prior to riser uncovery. The DHRS PSSR analysis is concerned with DHRS performance when the riser remains covered and the large majority of RCS flow occurs above the riser and across the entirety of the SG tube bank. The impact of riser holes on DHRS performance following riser uncovery is to allow for a small amount of continued RCS circulation This provides a slight improvement in heat transfer to the SG and is expected to have no impact on the figure of merit.
ER-P000-8076, "PRA Impacts Analysis of Boron Redistribution Design Changes"	• ECCS System Model – As a model simplification, CNV level sensor is the only sensor modeled for ECCS actuation. Sensor miscalibrations due to operator error were also considered as potential causes of spurious ECCS actuations. While normal operating conditions would meet the ECCS temperature interlock, the logic also requires two low RCS pressure and two high CNV pressure conditions to be met. Two out of four sensors have to be meet for ECCS actuation. A human mis-calibration of 1.7E-3, operator mis-calibrates SFMs in a group of four during test and maintenance. Compared to the likelihood of main valve leakage, the applicant concluded miscalibration of **both** conditions is a negligible contributor to spurious ECCS actuation. Furthermore, the IAB block valve design feature prevents the main ECCS valve during module operation [[ ]]. The IAB is implemented in the form of a normally open check valve in the trip line that closes to prevent actuation when the RPV to CNV differential pressure across an ECCS valve is above a predetermined value. • The MPS is a highly reliable system that incorporates redundancy in multiple areas of the architecture, including sensors and detectors. The MPS logic change of this signal is unchanged by the setpoint change so there is no impact on the ECCS fault tree. • This new actuation requires three sensors for ECCS actuation (RCS pressure, RCS temperature, and CNV pressure). Based on a review of the top 25 ECCS cutsets, sensor failures are shown to have a negligible impact on overall ECCS reliability.

Document	Summary of Document
	• Documents the event trees that have been modified due to the revised thermal-hydraulic analyses and revised success criteria.
"Audit information PRA plot request," May 5, 2020	Sensitivity study for CVCS charging line break outside containment with no DHRS, one train of DHRS, one train of DHRS until SG uncovered, both trains of DHRS available until SG uncovered. Plots were provided for following parameters: • DHRS energy removal vs time • DHRS condensate line mass flow rate vs time • SG collapsed level vs time • Downcomer collapsed level vs time • Riser collapsed level vs time • RCS pressure vs time • PCT vs time Sensitivity studies for ECCS actuation timing on LOCA inside the CNV with failure of the Containment Evacuation System (CES) to isolate with single RRV opening. Case is initiated with an inadvertent reactor vent valve opening. The CES line is assumed to fail closed and is modeled as a 2-inch diameter hole at the top of containment. Plots were provided for the following parameters: • CES Integrated Break Flow versus time • CNV Pressure versus time • CNV Pressure Blowdown versus time • RPV Pressure versus time • PCT versus time • Riser Level versus time • CNV Level versus time
"Audit information PRA plot request sensitivities," May 6, 2020	Additional sensitivity study plots for LOCA inside CNV with CES isolation failure. Two categories of sensitivities include 1) allowing all ECCS valves to open on ECCS signal assuming DHRS is unavailable and 2) enabling one train of DHRS with only one RRV opening. ECCS actuation timing varied as follows: baseline of 53 sec, 10 minutes, 20 minutes, 30 minutes, 40 minutes. Plots were provided for following parameters: • CES integrated break flow vs time • Riser level vs time • CNV level vs time

Document	Summary of Document
"SBLOCA ATWS plots for discussion," May 7, 2020	Summary of small break LOCA with ATWS simulations to demonstrate the effectiveness of the new ECCS actuation with smaller breaks sizes. The N-RELAP model employed BOC conditions with 1-D nodal kinetics and 2 dimensional nodalization for the riser and downcomer regions to allow mixing of fluids. The analyzed cases covered breaks in the high point vent line (steam space) and the RCS discharge line (liquid space). The break sizes analyzed included 100 percent, 7 percent, 2 percent, and 1 percent breaks. Plots were provided for following parameters: • Boron concentrations in the core, riser and downcomer vs time • RPV pressure vs time • Integrated RRV flow vs time • Riser hole flow vs time • Riser level vs time • Downcomer level vs time • PCT vs time • Core power vs time • Total reactivity vs time
"SB LOCA ATWS plots for discussion," May 14, 2020	Provides supplemental information regarding the small break LOCA ATWS simulations to support PRA audit of boron distribution design changes. The new information contains CNV level plots as well as trip timing tables to facilitate identification of which signal actuates the ECCS for each case.
"PRA Small Break LOCA ATWS Analysis," May 26, 2020	Documents the small break LOCA ATWS analysis information provided in the eRR. Provides additional details on the riser and downcomer nodalization and the kinetics model used.
"PRA Action Item CHF information," May 26, 2020	Provides response to staff question on what critical heat flux correlation is used for the rod heat structure in the LOCA ATWS analysis. [[ ]] is used.
"SBLOCA ATWS Supplemental Information," June 2, 2020	Provides response to staff question on power oscillations occurring prior to ECCS actuation for the discharge line break cases provided in the ERR May 26th. Specifically core power oscillations occurring to ECCS actuation are investigated. All cases discussed are discharge line breaks. The smaller breaks demonstrate core power oscillations in the moments that the liquid level reaches the top of the riser and continuous RCS circulation is interrupted. These power oscillations occur coincide with "burping" of flow over the top of the riser. The burping and power oscillations cease as level as the level continues to drop below the top of the riser.

Document	Summary of Document
	For the smallest breaks, where the level drops more slowly, the oscillations last longer and are more pronounced. Boron redistribution between the riser and the downcomer regions continues to occur during these oscillations and prior to ECCS actuation. A table is provided showing the difference in boron concentration between the core and lower downcomer decreasing with decreasing break sizes (15 percent, 7 percent, 4 percent, 2 percent, 1 percent). Plots were provided for following parameters: • Core and downcomer boron concentrations vs time • Liquid flow over the riser vs time • Core power vs time • PCT vs time
"PRA audit supplemental info draft for 6-12 event frequency," June 12, 2020	Provides applicant's estimated frequency of the LOCA ATWS scenario. Assumes that the frequency of CVCS charging line LOCA inside containment is approximately 1E-4 per year, and the failure probability of the reactor trip system is on the order of [[ ]]. The applicant notes that operator action to remove power from the reactor trip breakers would lower the frequency to [[ ]].
"PRA audit supplemental info draft for 6-12 kinetic parameters," June 12, 2020	Provides a description of the process used to develop the kinetics parameters used in the N-RELAP calculations.
"PRA Audit Info Slides 0612 Discussion," June 16, 2020	The first set of plots present NRELAP 5 modeling of boron transport in the LOCA ATWS analysis. These plots show that liquid and boron flow around the RCS is stable and remains in the positive direction for the relevant flow paths during the time period of riser uncovery and prior to ECCS actuation. Back and forth liquid oscillations are not observed between the riser and the downcomer. The licensee concludes that artificial boron mixing (i.e. driven by numerical flow oscillations) is not occurring The second set of plots demonstrates the potential consequences of a reactivity insertion upon ECCS actuation in the LOCA ATWS analysis. Plots are presented for the 4 percent break of the discharge line with nominal ECCS actuation setpoints as well as a conservative sensitivity case with delayed ECCS actuation ([[ ]] psia) and increased reactivity insertion due to increased dilution in the downcomer to demonstrate the large margin to fuel failure criteria. Supplemental reactivity hand calculation provided using the Fuchs Nordheim method showing large margin to the fuel failure criteria.

Document	Summary of Document
"NuScale response non-LOCA ATWS questions," June 17, 2020	Provides response to staff question on non-LOCA ATWS questions. The general transient ATWS event tree sequences are representative of a subset of event progressions in which the core could be either critical or subcritical at the time of ECCS actuation. CVCS injection will provide negative reactivity insertion because the injected coolant will be borated and the MPS sends a signal to isolate the demineralized water isolation valves on a reactor trip actuation. Void collapse in the core region is judged to not be credible based on the core mass of approximately [[ ]] and CVCS injection rate of [[ ]]. At this flow rate, the coolant injection in the riser will not rapidly reduce liquid temperatures in the core regions and will therefore not rapidly condense and collapse voids. There are several different failures that contribute to the probability of ATWS. These include the trip system failures as well as common cause failure of 3 or more control rods failing to insert. This event tree includes common cause failure of control rods failing to insert.
Assessment and Disposition of Geyser Instability for NuScale LTC Conditions 7/1/20	Flow Instability Assessment NuScale LTC conditions 1. Flow dynamics are driven by level differences not density differences (e.g. a manometer), so large flow incursions are not possible. 2. The subcooling in the riser is low. The low amount of subcooling results in a small condensation rate in the lower riser. 3. The core boiling is very low due to low decay heat levels. Therefore, the amount of phase change in the core/riser section is sufficiently low to keep the inventory variation minimal in one of the manometer legs.

TR-0516-49422, "Loss-Of-Coolant Accident Analysis Methodology" **(NRR/DSS/SNRB)**

The NRC staff audited the NuScale change packages for TR-0516-49422 (CP-2083 and CP-2107) and confirmed that the topical report would be updated appropriately. The NRC staff provided comments on the changes proposed largely based on its review of the updated LOCA spectrum calculation, EC-0000-2749. The NRC staff wanted to make sure that the new ECCS logic for the low pressure signal and the riser holes were adequately described and their overall relative effect on LOCA liquid and steam space breaks, as well as the IORV cases. The NRC

staff agreed that the LOCA example cases from EC-0000-4888 documented in the LOCA TR did not require updating for this design change.

Staff noted that the NRELAP5 base model (EC-A010-1782) was updated to revision 3 to incorporate these design changes.

TR-0516-49416, "Non-Loss-Of-Coolant Accident Analysis Methodology **(NRR/DSS/SNRB)**

The NRC staff audited document ECN-0000-8080, "Assessment of Riser Holes Impact on Non-LOCA Events," to understand the effect of riser holes on the non-LOCA topical report and Chapter 15 events analyzed using the non-LOCA methodology. The NRC staff noted the following:

- The purpose of the document was to assess the impact of the riser holes on non-LOCA event progressions.
- The document provided the NRELAP5 input to incorporate the riser holes, which was consistent with the description in EC-A010-1782, Revision 3.
    - The riser hole form loss applied was [[ ]].
- The applicant analyzed five transient cases using NRELAP5, one representative case from each event category, with and without riser holes.
    - The cases chosen were the limiting cases for the selected events with respect to the associated figures of merit listed in the table below.
    - The steady-state runs showed very little difference when the riser holes were modeled.
        - The largest change was that RCS flow increased by about [[ ]] with riser holes modeled.
    - The applicant provided select transient plots that showed the transient progressions changed minimally.
        - RCS average temperature was slightly reduced when riser holes were modeled due to additional flow to the SGs.
        - MPS signal timing changed slightly [[ ]] in some cases.
    - The table below provides the sensitivity results. As can be seen, the riser holes' impact to figures of merit is negligible.
- Based on the insignificant impacts, the applicant deemed further evaluations, including those involving downstream methodologies, unnecessary.

Event	Figure of Merit	Without Riser Holes	With Riser Holes
15.1.1, Decrease in Feedwater Temperature	NRELAP5-predicted MCHFR	[[ ]]	[[ ]]

15.2.7, Loss of Normal Feedwater	Peak RCS pressure (psia)	[[ ]]	[[ ]]
15.4.2, Uncontrolled Rod Withdrawal at Power	Peak reactor power (percent)	[[ ]]	[[ ]]
	NRELAP5-predicted MCHFR	[[ ]]	[[ ]]
15.5.1, CVCS Malfunction	Peak SG pressure (psia)	[[ ]]	[[ ]]
15.6.3, SG Tube Failure	Mass release (lbm)	[[ ]]	[[ ]]

In addition, the NRC staff audited the NuScale change package for TR-0516-49416 (CP-2084) and confirmed that the topical report would be updated appropriately.

8.0 EXIT BRIEFING

The NRC staff conducted an audit exit meeting via bridge line on June 26, 2020. During the meeting, the NRC staff reiterated the purpose of the audit and discussed the audit activities and expected outcome. References to the detailed questions are provided in Section 11 of this audit summary.

In summary, the NRC staff examined several technical reports, calculation notes, and supporting documents in the eRR and held daily teleconferences to discuss various topics and issues as they emerged and developed during the audit. In general, the NRC staff gained significant insight regarding the three NuScale design changes as they relate to the boron redistribution issue. With NuScale's support and cooperation during the audit, the NRC staff was able to resolve many technical issues.

9.0 OPEN ITEMS AND PROPOSED CLOSURE PATHS

Not applicable.

10.0 DEVIATIONS FROM THE AUDIT PLAN

The audit was originally established to review changes made to instrumentation and controls setpoints and/or logic to address an issue related to boron redistribution and the associated return to power analyses associated with Chapter 15, Section 15.0.6, of the FSAR. During the audit review process, NuScale added information for two additional design changes to the eRR, which significantly expanded the scope of review.

11.0 REFERENCES

1. Audit Plan for the Regulatory Audit of the NuScale Power, LLC DCA, Chapter 6, "Engineered Safety Features," Chapter 7, "Instrumentation and Controls," and Chapter 15, "Transient and Accident Analysis," related to change in instrumentation and Controls Setpoints and/or Logic, issued March 2, 2020 (ADAMS Accession No. ML20059N687).